ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Rail Alignment Tolerance Requirements **72.48 Log No.: SE00121**

D3-HSM-43; 118/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: J.A. CRUNKLETON JA Crunkleton
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-137 Date: 12·3·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 12-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 12-3-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC/SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Rail Alignment Tolerance Requirements			72.48 Log No.: SE00121	
D3-HSM-43; 118/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) rail alignment tolerance requirements.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 4.3, 5.1, 7.3, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Rail Alignment Tolerance Requirements **72.48 Log No.: SE00121**

D3-HSM-43; 118/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 3 of 5**

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of the rail alignment tolerance requirements design change. The subject activity clarifies the HSM-DSC rail alignment requirements. The rails shall be level within 1/16" between the front and rear of each module. Additionally, the two rails in each module shall not deviate by more than 1/16" in elevation when measured across the rails. Tightening the installation tolerances improves the rail alignment and overall safety of the system, as well as provide for a smooth transfer of the DSC. This design change is an improvement to the HSM which does not adversely affect the HSM design or analysis. Since 1993, all fuel moves have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the rails of the HSM.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Rail Alignment Tolerance Requirements **72.48 Log No.: SE00121**

D3-HSM-43; 118/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 4 of 5**

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity clarifies the HSM-DSC rail alignment requirements. The rails shall be level within 1/16" between the front and rear of each module. Additionally, the two rails in each module shall not deviate by more than 1/16" in elevation when measured across the rails. Tightening the installation tolerances improves the rail alignment and overall safety of the system, as well as provide for a smooth transfer of the DSC. This design change is an improvement to the HSM which does not adversely affect the HSM design or analysis. Since 1993, all fuel moves have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety, and will not create the possibility of a new malfunction not previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. After a thorough and intense review, it was concluded that this activity would not create the possibility of a new accident not previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a rail alignment tolerance requirements design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The rail alignment tolerance requirements design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Rail Alignment Tolerance Requirements **72.48 Log No.: SE00121**

D3-HSM-43; 118/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) rail alignment tolerance requirements.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM W8x48 Beam Orientation **72.48 Log No.: SE00122**

D3-HSM-44; 119/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 **SIGNATURE / DATE**	11/10/97 **SIGNATURE / DATE**	11/11/97 **SIGNATURE / DATE**

[✓ Approved] Disapproved
Signature: J.A. CRUNKLETON JA Crunkleton
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for [✓ **GS-DES**], **GS-TES, or PE-PDSU**
MICHAEL J. GAHAN III
Date 11-14-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-137 Date: 12-3-97

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 12-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 12-3-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No ✓

Signature: ________ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: .:________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM W8x48 Beam Orientation **72.48 Log No.: SE00122**

D3-HSM-44; 119/129; ES199601368 Supplement 001 Revision 0000 Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) W8x48 beams orientation.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 4.3, 5.1, 7.3, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM W8x48 Beam Orientation			72.48 Log No.: SE00122	
D3-HSM-44; 119/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of the W8x48 beam orientation design change. The subject activity provided an allowance for the orientation of the W8x48 beams to be reversed (180° in the horizontal plane) to match the as-built centerline of the HSM access sleeve. It also allowed the length of the slots to be increased if required to match the as-built location of the rails. This design change did not affect the structural adequacy of the beams in any way, in that the strong axis remained in the vertical plane. The intent of this design change was to give the field personnel additional flexibility in the construction of the DSC support structure. Since 1993, all fuel moves have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the support beams of the HSM.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM W8x48 Beam Orientation			72.48 Log No.: SE00122	
D3-HSM-44; 119/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity provided an allowance for the orientation of the W8x48 beams to be reversed (180° in the horizontal plane) to match the as-built centerline of the HSM access sleeve. It also allowed the length of the slots to be increased if required to match the as-built location of the rails. This design change did not affect the structural adequacy of the beams in any way, in that the strong axis remained in the vertical plane. The intent of this design change was to give the field personnel additional flexibility in the construction of the DSC support structure. Since 1993, all fuel moves have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety, and will not create the possibility of a new malfunction not previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. After a thorough and intense review, it was concluded that this activity would not create the possibility of a new accident not previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a W8x48 beam orientation design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The W8x48 beam orientation design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM W8x48 Beam Orientation			72.48 Log No.: SE00122	
D3-HSM-44; 119/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) W8x48 beams orientation.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Lifting Lugs Hole Diameter Design Change **72.48 Log No.: SE00123**

D3-HSM-45; 120/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 **SIGNATURE / DATE**	11/10/97 **SIGNATURE / DATE**	11/11/97 **SIGNATURE / DATE**

[✓ Approved] Disapproved

Signature: J.A. CRUNKLETON JA Crunkleton
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for **GS-DES, GS-TES, or PE-PDSU**
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-137 Date: 12-3-97

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 12-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 12-3-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: ________ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ________________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Lifting Lugs Hole Diameter Design Change			72.48 Log No.: SE00123	
D3-HSM-45; 120/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) door lifting lugs.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 4.3, 5.1, 7.3, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Lifting Lugs Hole Diameter Design Change	**72.48 Log No.: SE00123**
D3-HSM-45; 120/129; ES199601368 Supplement 001 Revision 0000	**Page 3 of 5**

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of the door lifting lug hole diameter design change. The subject activity revised the hole diameter in the HSM door lifting lugs from 11/16" to 1.0" to allow the use of 3/4" shackle pins. Per the AISC Eight Edition, Section 1.16.5, the minimum edge distance from the center of a 1.0" hole to a rolled edge is 1-1/4", and that distance requirement was still met through this design change. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the door of the HSM.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Lifting Lugs Hole Diameter Design Change			72.48 Log No.: SE00123	
D3-HSM-45; 120/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity revised the hole diameter in the HSM door lifting lugs from 11/16" to 1.0" to allow the use of 3/4" shackle pins. Per the AISC Eight Edition, Section 1.16.5, the minimum edge distance from the center of a 1.0" hole to a rolled edge is 1-1/4", and that distance requirement was still met through this design change. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety, and will not create the possibility of a new malfunction not previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. After a thorough and intense review, it was concluded that this activity would not create the possibility of a new accident not previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a door lifting lug hole diameter design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The door lifting lug hole diameter design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Lifting Lugs Hole Diameter Design Change **72.48 Log No.: SE00123**

D3-HSM-45; 120/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) door lifting lugs.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Weld Requirements **72.48 Log No.: SE00124**

D3-HSM-46; 121/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: J.A. CRUNKLETON
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-137 Date: 12-3-97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 12-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 12-3-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Weld Requirements			**72.48 Log No.: SE00124**	
D3-HSM-46; 121/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 2 of 5**

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) door frame weld requirements.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 4.3, 5.1, 7.3, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Weld Requirements			**72.48 Log No.: SE00124**	
D3-HSM-46; 121/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 3 of 5**

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of the door frame weld requirements design change. The subject activity added a requirement for a seal weld between structural stitch welds between the door frame support angles and the HSM embed plates. This change was made to prevent water seepage between the angle and the embed plate of the door frame. Seal welds are considered non-structural and are added to improve the seal of affected areas. This change is an improvement and does not affect the HSM design or analysis. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the door of the HSM.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Weld Requirements **72.48 Log No.: SE00124**

D3-HSM-46; 121/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity added a requirement for a seal weld between structural stitch welds between the door frame support angles and the HSM embed plates. This change was made to prevent water seepage between the angle and the embed plate of the door frame. Seal welds are considered non-structural and are added to improve the seal of affected areas. This change is an improvement and does not affect the HSM design or analysis. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety, and will not create the possibility of a new malfunction not previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. After a thorough and intense review, it was concluded that this activity would not create the possibility of a new accident not previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a door frame weld requirements design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The door frame weld requirements design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Weld Requirements	**72.48 Log No.: SE00124**
D3-HSM-46; 121/129; ES199601368 Supplement 001 Revision 0000	**Page 5 of 5**

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) door frame weld requirements.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM DSC Seismic Restraint Design Change **72.48 Log No.: SE00125**

D3-HSM-47; 122/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: J.A. CRUNKLETON JA Crunkleton
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for [✓ GS-DES] GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-137 Date: 12-3-97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 12-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 12-3-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: M Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: .:______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM DSC Seismic Restraint Design Change **72.48 Log No.: SE00125**

D3-HSM-47; 122/129; ES199601368 Supplement 001 Revision 0000 Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) seismic restraint.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 4.3, 5.1, 7.3, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM DSC Seismic Restraint Design Change			72.48 Log No.: SE00125	
D3-HSM-47; 122/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of the seismic restraint design change. The subject activity revised the DSC seismic restraint to minimize its weight and to add a handle for remote installation. These changes will result in reduced occupational exposure during the installation of the restraint. The structural adequacy of the revised restraint is verified in calculation BGE001.0218, "Revised DSC Seismic Restraint". Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the seismic restraint of the HSM.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM DSC Seismic Restraint Design Change **72.48 Log No.: SE00125**

D3-HSM-47; 122/129; ES199601368 Supplement 001 Revision 0000

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity revised the DSC seismic restraint to minimize its weight and to add a handle for remote installation. These changes will result in reduced occupational exposure during the installation of the restraint. The structural adequacy of the revised restraint is verified in calculation BGE001.0218, "Revised DSC Seismic Restraint". Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety, and will not create the possibility of a new malfunction not previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. After a thorough and intense review, it was concluded that this activity would not create the possibility of a new accident not previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a seismic restraint design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The seismic restraint design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM DSC Seismic Restraint Design Change			**72.48 Log No.: SE00125**	
D3-HSM-47; 122/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 5 of 5**

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) seismic restraint.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM W8x48 Beam Connection Design Change **72.48 Log No.: SE00126**

D3-HSM-48; 123/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye	Resp. Indv.: C. J. Dobry	Resp. Indv.: R. H. Beall
Work Group: Licensing	Work Group: PES	Work Group: NFM
Getachew Tesfaye 11/12/97	11/10/97	11/11/97
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: J.A. CRUNKLETON J A Crunkleton Jr
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan for
GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-137 Date: 12·3·97

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 12-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 12-3-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: ________ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: .: ________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM W8x48 Beam Connection Design Change			72.48 Log No.: SE00126	
D3-HSM-48; 123/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) W8x48 beams connections.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 4.3, 5.1, 7.3, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM W8x48 Beam Connection Design Change	**72.48 Log No.: SE00126**
D3-HSM-48; 123/129; ES199601368 Supplement 001 Revision 0000	**Page 3 of 5**

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of the W8x48 beam connection design change. The subject activity added slots to the W8x48 cross-beam supports to provide more flexibility in the installation of the support steel. In addition, the rail support steel was mounted on 10-1/2" x 9" x 3/4" plates and attached to the cross beams by 4 - 5/8" diameter bolts. The bolts were used to add flexibility to the joints to eliminate thermal stresses due to differential movement of the beams. The mounting bolts provide an equivalent shear resistance to the welds which were previously designed. This change also simplifies the installation and alignment of the DSC rails. The intent of this design change was to give the field personnel additional flexibility in the construction of the DSC support structure and to eliminate the thermal stresses. Since 1993, all fuel moves have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the beams of the HSM.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity added slots to the W8x48 cross-beam supports to provide more flexibility in the installation of the support steel. In addition, the rail support steel was mounted on 10-1/2" x 9" x 3/4" plates and attached to the cross beams by 4 - 5/8" diameter bolts. The bolts were used to add flexibility to the joints to eliminate thermal stresses due to differential movement of the beams. The mounting bolts provide an equivalent shear resistance to the welds which were previously designed. This change also simplifies the installation and alignment of the DSC rails. The intent of this design change was to give the field personnel additional flexibility in the construction of the DSC support structure and to eliminate the thermal stresses. Since 1993, all fuel moves have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety, and will not create the possibility of a new malfunction not previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. After a thorough and intense review, it was concluded that this activity would not create the possibility of a new accident not previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a W8x48 beam connection design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The W8x48 beam connection design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM W8x48 Beam Connection Design Change				72.48 Log No.: SE00126
D3-HSM-48; 123/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) W8x48 beams connections.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Beam Support Angle Change **72.48 Log No.: SE00127**

D3-HSM-49; 124/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES	Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	R. H. Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: J.A. CRUNKLETON JA Crunkleton
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for [✓ GS-DES], GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-137 Date: 12-3-97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 12-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 12-3-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: JR Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Beam Support Angle Change			72.48 Log No.: SE00127	
D3-HSM-49; 124/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) W8x48 beams support angles.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 4.3, 5.1, 7.3, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Beam Support Angle Change	**72.48 Log No.: SE00127**
D3-HSM-49; 124/129; ES199601368 Supplement 001 Revision 0000	**Page 3 of 5**

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of the W8x48 beam support angle design change. The subject activity increased the size of the support angles for the cross beam supports from L4X3 to L4X4. This design change was made to meet the AISC requirement to provide minimum edge distances for the attachment bolts. The slightly larger angle provides a leg length of 4" in lieu of 3", which increased the edge distance from 1/2" to 1-1/2". The AISC required edge distance for a 5/8" bolt is 7/8". Since 1993, all fuel moves have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the beams of the HSM.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Beam Support Angle Change **72.48 Log No.: SE00127**

D3-HSM-49; 124/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity increased the size of the support angles for the cross beam supports from L4X3 to L4X4. This design change was made to meet the AISC requirement to provide minimum edge distances for the attachment bolts. The slightly larger angle provides a leg length of 4" in lieu of 3", which increased the edge distance from 1/2" to 1-1/2". The AISC required edge distance for a 5/8" bolt is 7/8". Since 1993, all fuel moves have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety, and will not create the possibility of a new malfunction not previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. After a thorough and intense review, it was concluded that this activity would not create the possibility of a new accident not previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a W8x48 beam support angle design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The W8x48 beam support angle design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Beam Support Angle Change	**72.48 Log No.: SE00127**
D3-HSM-49; 124/129; ES199601368 Supplement 001 Revision 0000	**Page 5 of 5**

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) W8x48 beams support angles.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Beam Support Steel Weld Changes **72.48 Log No.: SE00128**

D3-HSM-50; 125/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: J.A. CRUNKLETON J A Crunkleton
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-137 Date: 12-3-97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 12-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 12-3-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Beam Support Steel Weld Changes				**72.48 Log No.: SE00128**
D3-HSM-50; 125/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 2 of 5**

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) W8x48 beams support steel welds.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 4.3, 5.1, 7.3, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Beam Support Steel Weld Changes **72.48 Log No.: SE00128**

D3-HSM-50; 125/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of the W8x48 beam support steel weld design change. The subject activity changed the welds attaching the rail support steel to the cross beams mounting plates to change from one 1/2" and one 1/4" groove weld to two 3/8" groove welds. The new weld configuration has an identical 3/4" total throat as the original design, and as such, the allowable shear resistance for welds will remain the same. Since 1993, all fuel moves have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

The subject design change allows the"

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the beams of the HSM.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Beam Support Steel Weld Changes **72.48 Log No.: SE00128**

D3-HSM-50; 125/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity changed the welds attaching the rail support steel to the cross beams mounting plates to change from one 1/2" and one 1/4" groove weld to two 3/8" groove welds. The new weld configuration has an identical 3/4" total throat as the original design, and as such, the allowable shear resistance for welds will remain the same. Since ISFSI loading operations began in November of 1993, all fifteen fuel moves to date have resulted in a smooth transfer of the DSC from the TC into the HSM. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety, and will not create the possibility of a new malfunction not previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. After a thorough and intense review, it was concluded that this activity would not create the possibility of a new accident not previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a W8x48 beam support steel weld design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The W8x48 beam support steel weld design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Beam Support Steel Weld Changes **72.48 Log No.: SE00128**

D3-HSM-50; 125/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) W8x48 beams support steel welds.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Design Change No. 1 **72.48 Log No.: SE00129**

D3-HSM-51; 126/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye	Resp. Indv.: C. J. Dobry	Resp. Indv.: R. H. Beall
Work Group: Licensing	Work Group: PES	Work Group: NFM
11/12/97	11/10/97	11/4/97
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: J.A. CRUNKLETON JACrunkleton
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for ~~GS-DES~~, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-137 Date: 12-3-97

Recommend Approval ✓ Recommend Disapproval ____

Signature: ____ Date 12-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved ____

Signature: ____ Date 12-3-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ____ No X

Signature: ____ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ____

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Design Change No. 1 **72.48 Log No.: SE00129**

D3-HSM-51; 126/129; ES199601368 Supplement 001 Revision 0000 Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) door frame angles.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 4.3, 5.1, 7.3, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Design Change No. 1			72.48 Log No.: SE00129	
D3-HSM-51; 126/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of the door frame angle design change number 1. The subject activity changed the four door guide angles from L7x4x¾ with L9x4x5/8. This design change is analyzed in calculation BGE001.0213 Revision 2, HSM Door Analysis. The reason for the change was to provide an angle with a longer leg (9" in lieu of 7") which would provide the required angle overlap and length to serve as a guide for the HSM door, which is 12¾" thick. This angle, combined with a L7x4x¾ and the required angle overlap, results in a guide spacing of 13-1/8". The calculation verified that the stresses associated with the angle change were safely below the allowables. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the door of the HSM.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Design Change No. 1 **72.48 Log No.: SE00129**

D3-HSM-51; 126/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity changed the four door guide angles from L7x4x¾ with L9x4x5/8. This design change is analyzed in calculation BGE001.0213 Revision 2, HSM Door Analysis. The reason for the change was to provide an angle with a longer leg (9" in lieu of 7") which would provide the required angle overlap and length to serve as a guide for the HSM door, which is 12¾" thick. This angle, combined with a L7x4x¾ and the required angle overlap, results in a guide spacing of 13-1/8". The calculation verified that the stresses associated with the angle change were safely below the allowables. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety, and will not create the possibility of a new malfunction not previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. After a thorough and intense review, it was concluded that this activity would not create the possibility of a new accident not previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a door frame angle design change number 1. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The door frame angle design change number 1 does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Design Change No. 1	**72.48 Log No.: SE00129**
D3-HSM-51; 126/129; ES199601368 Supplement 001 Revision 0000	**Page 5 of 5**

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) door frame angles.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Design Change No. 2 **72.48 Log No.: SE00130**

D3-HSM-52; 127/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11·7·97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 **SIGNATURE / DATE**	11/10/97 **SIGNATURE / DATE**	11/11/97 **SIGNATURE / DATE**

[✓ Approved] Disapproved

Signature: J.A. CRUNKLETON J.A. Crunkleton
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for ~~GS-DES~~, ~~GS-TES~~, **or PE-PDSU**
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-137 Date: 12·3·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 12-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 12-3-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Design Change No. 2 **72.48 Log No.: SE00130**

D3-HSM-52; 127/129; ES199601368 Supplement 001 Revision 0000 Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) door frame angles.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 4.3, 5.1, 7.3, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Design Change No. 2 **72.48 Log No.: SE00130**

D3-HSM-52; 127/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of the door frame angle design change number 2. The subject activity provided a construction alternative to substitute L8x4x¾ angles for all L7x4x¾ and L9x4x5/8 door frame angles. This design change is analyzed in calculation BGE001.0213 Revision 2, HSM Door Analysis. The reason for the change was to provide the constructor with one angle size for an order in lieu of two, and to eliminate the possibility of incorrectly constructing the frame by mixing the angles. This alternative design is equivalent to the original design, which was discussed in ISFSI Safety Evaluation SE00129. The use of this angle, along with the required angle overlap, results in a guide spacing of 13-1/8", which provides the spacing necessary for the 12¾" thick HSM door. The calculation verified that the stresses associated with the angle change were safely below the allowables. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the door of the HSM.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Design Change No. 2 **72.48 Log No.: SE00130**

D3-HSM-52; 127/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 4 of 5**

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity provided a construction alternative to substitute L8x4x¾ angles for all L7x4x¾ and L9x4x5/8 door frame angles. This design change is analyzed in calculation BGE001.0213 Revision 2, HSM Door Analysis. The reason for the change was to provide the constructor with one angle size for an order in lieu of two, and to eliminate the possibility of incorrectly constructing the frame by mixing the angles. This alternative design is equivalent to the original design, which was discussed in ISFSI Safety Evaluation SE00129. The use of this angle, along with the required angle overlap, results in a guide spacing of 13-1/8", which provides the spacing necessary for the 12¾" thick HSM door. The calculation verified that the stresses associated with the angle change were safely below the allowables. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety, and will not create the possibility of a new malfunction not previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. After a thorough and intense review, it was concluded that this activity would not create the possibility of a new accident not previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a door frame angle design change number 2. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The door frame angle design change number 2 does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Design Change No. 2			72.48 Log No.: SE00130	
D3-HSM-52; 127/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) door frame angles.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Welds Design Changes **72.48 Log No.: SE00131**

D3-HSM-53; 128/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11-7-97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97	11/10/97	11/11/97
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: J.A. CRUNKLETON JA Crunkleton
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-137 Date: 12·3·97

Recommend Approval ✓ Recommend Disapproval _______ Signature: _______ Date 12-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved _______ Signature: _______ Date 12-3-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes _______ No X

Signature: _______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: _______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Welds Design Changes **72.48 Log No.: SE00131**

D3-HSM-53; 128/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 2 of 5**

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) door frame angle welds.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 4.3, 5.1, 7.3, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Welds Design Changes			**72.48 Log No.: SE00131**	
D3-HSM-53; 128/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 3 of 5**

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of the door frame angle weld design change. The subject activity reduced the amount of weld used to attach the door frame angles to the embedded steel. This change was made to reduce the heat input to the concrete and lower the potential for concrete cracking. This design change is analyzed in calculation BGE001.0213, Revision 2, HSM Door Analysis. As shown in this calculation, the revised weld design meets the code allowables. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

The subject design change

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the door of the HSM.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the HSM which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Welds Design Changes **72.48 Log No.: SE00131**

D3-HSM-53; 128/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity reduced the amount of weld used to attach the door frame angles to the embedded steel. This change was made to reduce the heat input to the concrete and lower the potential for concrete cracking. This design change is analyzed in calculation BGE001.0213, Revision 2, HSM Door Analysis. As shown in this calculation, the revised weld design meets the code allowables. Based on this information, the subject design change will not affect the form, fit or function of the HSM, is not detrimental to the structural integrity of the HSM, and will not adversely affect the ability of the HSM to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety, and will not create the possibility of a new malfunction not previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. After a thorough and intense review, it was concluded that this activity would not create the possibility of a new accident not previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a door frame angle weld design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The door frame angle weld design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - HSM Door Frame Angle Welds Design Changes			**72.48 Log No.: SE00131**	
D3-HSM-53; 128/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 5 of 5**

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the HSM (Horizontal Storage Module) door frame angle welds.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Analysis of DSC under Postulated Cask Drop Accident **72.48 Log No.:SE00133**

ES199601368 Supplement 002 Revision 0000 **Page 1 of 8**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions or Bases?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: David A. Scheetz
PRINTED NAME AND SIGNATURE
Department: Duke Engineering & Services Date: 2-16-98

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
2/16/98	2/16/98	2/16/98
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: for J. REEVES 2/16/98
INDEPENDENT REVIEWER
Date 2/16/98

[✓ Approved] Disapproved
Signature: Michael J. Gahan
2/16/98 CCSD
GS-DES, GS-TES, or PE-PDSU
Date 2.17.98

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 98-018 Date: 2.17.98

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 2-17-98
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 2/17/98
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No ✓

Signature: JR Lemons Date: 5/15/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Analysis of DSC under Postulated Cask Drop Accident **72.48 Log No.:SE00133**

ES199601368 Supplement 002 Revision 0000 **Page 2 of 8**

Proposed Activity: This safety evaluation is prepared to clarify and correct the licensing basis for the NUHOMS system in use at the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) with respect to the postulated transfer cask drop accident. The USAR will be changed to properly account for the behavior of DSC components, and to correct the stress values and deflections that might be expected in the unlikely event of a cask vertical drop accident. This activity applies to canisters R008, R009, and R018-R024.

Change the Calvert Cliffs ISFSI USAR as follows:

1) USAR Section 8.2.5.2 qualifies the transfer cask for a cask drop accident by reference to "...the analytical methods presented in Section 8.2.5.2 of (the Topical Report)..." The USAR will be revised here to add a reference to, and a brief description of, BGE analyses which correct the analysis of the postulated transfer cask vertical drop accident in the Topical Report. These new analyses acknowledge that the connection between the DSC guide sleeves and the bottom spacer disk may fail at a higher vertical acceleration than the design previously anticipated, and the bottom spacer disk behavior has consequently been re-analyzed for this loading due to the guide sleeves. The support rods have also been reanalyzed for the effects of this new loading, including the use of ASME Appendix F in determining support rod allowable stress. No change to the USAR is anticipated for any other drop events.
2) USAR Tables 8.2-6 and 8.2-10 will be modified to show the new stress levels in the DSC components resulting from the transfer cask vertical drop.
3) USAR Section 8.2.5.2 specifies the maximum spacer disk deflection which might be expected under a cask drop accident of any orientation. This section will be updated.
4) USAR Sections 8.1.1, 8.1.2, and 8.2 will have a reference to the re-analysis added.

Reason for Activity:

An independent design review and assessment of TN-West (formerly known as Vectra Technologies, Inc., Pacific Nuclear, and NUTECH), performed by LES and commissioned by GPU Nuclear, Inc., identified a design issue with the generic NUHOMS-24P DSC system. The Calvert Cliffs ISFSI uses a similar NUHOMS-24P model, and is affected by this issue.

The issue involves the postulated transfer cask drop event, which is described in the ISFSI USAR Section 8.2.5. The ISFSI USAR concludes that a vertical , horizontal, or corner drop would:

- not affect the ability to retrieve fuel from the DSC stored inside the transfer cask
- not affect fuel cladding integrity
- not affect DSC structural integrity

The transfer cask drop analysis (which is not described in the ISFSI USAR, but is found in the DSC Structural Calculation C-91-076, Rev. 001, and the Topical Report for NUHOMS-24P) assumed, during a postulated vertical cask drop event, that the welds, which attach angle clips from the DSC bottom spacer disk to the DSC guide sleeves, will fail at 35 g's (SER p. 2-48). At that point, the guide sleeves would cease to exert any force on the bottom spacer disk. The independent design review and assessment concluded that the approach used to calculate the weld failure point was not conservative , and that the possibility exists that the welds may not fail until a higher acceleration level (> 35 g's) is reached. If the welds and angle clips were to remain intact at higher g levels during a vertical cask drop event, then more of the load associated with the deceleration of the DSC guide sleeves would be transferred to the DSC bottom spacer disk and four DSC support rods. As a result, the behavior of these components will change, and the stresses associated with the ISFSI USAR transfer cask drop event will slightly increase.

These concerns have been documented in BGE Issue Report IR1-011-183.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Analysis of DSC under Postulated Cask Drop Accident **72.48 Log No.:SE00133**

ES199601368 Supplement 002 Revision 0000 **Page 3 of 8**

Function(s) of affected SSC: Summarizing information provided in Chapter 4 of the USAR and ANSI-57.9, the dry shielded canister has containment, shielding, criticality control, configuration control related to fuel retrievability, and thermal safety functions. The primary function of the DSC is to provide containment for the spent nuclear fuel. This is achieved by the stainless steel shell and two inner cover plates (top and bottom ends) which are welded to the shell assembly. There are also outer cover plates (top and bottom) to further assure containment integrity. These are integral with the shield plugs. The DSC provides gamma shielding at its ends by the use of lead shield plugs. These provide ALARA dose rates at the top of the canister during drying and sealing operations and at the bottom for minimizing dose rates during DSC to HSM loading operations and at the HSM door during storage.

Criticality and configuration control is provided by the DSC's internal basket assembly. A series of nine spacer discs and four axial support rods maintain the fuel assemblies in known positions under all normal and accident conditions. The thickness and location of the spacer discs, plus the relative location of the fuel assemblies and the DSC basket material achieve the criticality and configuration control functions. The DSC maintains the helium cover gas which is required for heat rejection and corrosion control. Heat is transferred via thermal radiation and conduction from the fuel through the spacer discs and cover gas to the DSC shell, where it is convectively cooled during HSM interim storage. The presence of helium and spacer discs achieves the heat transfer function.

The on-site transfer cask provides shielding during the DSC closure and transfer operations. The transfer cask also provides structural protection for the DSC against natural and operation hazards during the transfer operations and loading of the DSC into the HSM for interim storage.

ISFSI USAR Revision No.: 6

ISFSI USAR Sections Reviewed: The main chapters reviewed were 3, 4, 5, 7, and 8. The key sections reviewed were 3.6, 4.2, 4.7, 5.1, 7.4, 7.5, 7.6, 8.1, and 8.2.

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ YES ☒ NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity.

Critical functions that must be maintained are containment, shielding, criticality control, configuration control related to fuel retrievability, and thermal safety functions. The cask drop scenario imposes limiting stresses on the transfer cask and the DSC components. Stresses beyond code allowables or unacceptable deflections of safety related DSC and transfer cask equipment/materials define the malfunctions needing to be considered when evaluating this activity involving the drop scenario. The activity in no way degrades the reliability of components important to safety since the design of the DSC is unchanged by this activity. There is no change to the design or operation of the NUHOMS system caused by this activity. While there are slight increases in stress levels of components of the DSC, the stresses are still below allowable values. Hence, there is no increase in the probability of malfunctions resulting from the accident.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Analysis of DSC under Postulated Cask Drop Accident **72.48 Log No.:SE00133**

ES199601368 Supplement 002 Revision 0000 **Page 4 of 8**

☐ YES ☒ NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity.

The potential malfunctions that could result from this change are the excessive stresses and permanent deformation of the bottom spacer disk, support rods, and/or guide sleeves which could adversely affect canister containment capability. Any breach in this containment capability could have direct radiological consequences, in the form of increased dose rates. Analyses using elastic-plastic methods consistent with the original bilinear elastic-plastic model (SER p. 2-48) have determined that increases in stress and strain levels in the subject components will occur. However, the stress levels are still below the allowable values and will not affect this containment capability (Ref. 6). Deformations are addressed below. For the case of the support rods, introduction of combined applied compression and bending loads from the bottom spacer disk, which was not previously adequately considered, introduces a new load case for the support rods. Again, analysis has shown this load case to be acceptable, with stresses below allowable values (Ref. 6). All other structural elements of the canister have been considered in the analysis, and the analysis conservatively envelopes the use of both stainless and carbon steel materials for the spacer disks. The stress levels indicate that there is no effect on the containment or structural integrity of the DSC. Hence, the radiological consequences of any component malfunction associated with the top cask drop are not increased.

In a cask bottom drop accident scenario, the gap between the bottom of the guide sleeves (connected to the bottom spacer disk), and the bottom cover plate would close well before the connection would fail. This is consistent with existing Topical Report p. 8.2-35.

Criticality control is assured as part of the existing design basis and technical specifications by the physical properties and history of the fuel, by mechanical control of the assemblies' locations in the DSC basket, by neutron absorption of the materials of the basket, by Calvert Cliffs administrative controls over fuel identification and handling, and by the presence of soluble boron in the fuel pool for wet operations (Ref. 1). The maintenance of continued positive control over criticality has been examined as part of this evaluation. In the event of a cask end drop, section 3.3.4 of the USAR has shown that introduction of a moderator would be necessary to cause a criticality concern. The violation of DSC structural integrity that would be necessary to allow the introduction of that moderator has been shown to be incredible above. None the less, a criticality analysis of the effects of spacer disk in-plane deformations on guide sleeve center-to-center spacing has been performed, and has determined that criticality control is still maintained over the fuel in the DSC in the event of a cask end drop (Ref. 13).

The Technical Specification requires retrievability of fuel from the DSC for any drop greater than 15". Analyses have determined the maximum guide sleeve deflection which might be expected from the cask vertical drop accident in the plane of the bottom spacer disk at the guide sleeve connection point. The permanent in-plane deflection is predominantly due to local deformation at the clip connection point to the guide sleeve (Ref. 6), and can directly impact retrievability. This deformation typically does not exceed the gap between the guide sleeve and the lower spacer grid of the fuel assembly, which is nominally computed to be one-half of 0.52 inches (Ref. 14), or 0.26 inches on each side, at this location. However, under certain circumstances this deformation may exceed the gap and "pinch" the fuel assembly. A small nominal force has been computed to be necessary (Ref. 6) to extract the fuel assembly from the DSC if this "pinched" condition should occur. The effects on the lower spacer grid, on the stresses in the fuel assembly guide tubes and upper end fitting posts, and on the fuel handling machine hoist during retrieval have been examined. The force on the lower spacer grid is comparable to the self weight that a spacer grid normally supports in the horizontal orientation, and hence is not a concern; The fuel assembly components are so rigid in the axial direction that any stress increases are inconsequential; Finally, the overload

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setting of the spent fuel handling machine hoist is procedurally verified to be much greater than the assembly weight plus the extraction force required (Ref. 15). The effects of the 75 g force acting on the spacer disk itself, together with any effect on other spacer disks, have been considered (Ref. 6), and do not impact the above-described "dimpling" determination or cause guide sleeve deflections which permanently close the guide sleeve-fuel assembly gap at spacer disks other than the bottom spacer disk. Hence, retrievability is assured.

☐ YES ☒ NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: The probability of occurrence of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity.

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. Of those accidents discussed in the SAR, only the cask drop scenario is affected. The probability of occurrence is only dependent on the drop initiating event frequency. There is no change to the design or operation of the NUHOMS system caused by this activity, or to the drop initiating event frequency. Therefore, the probability of occurrence of the cask drop is not changed. The SER (p. 2-43) states that the seismic analyses of certain components of the DSC were based upon scaling the results of the vertical cask drop analysis. However, examination of the design basis reveals that the seismic analyses of those components whose behavior will differ because of this activity, the bottom spacer disk and support rods, did not utilize any of the results of the drop analysis, instead relying for their design on the deadweight analysis. Therefore, there is no change in the probability of occurrence of any analyzed accident.

☐ YES ☒ NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident: The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity.

The USAR reports that an accidental cask drop results in the maintenance of continued structural and containment integrity. However, the computation of dose rates resulting from the cask drop is conservatively based on a total loss of solid neutron shielding in the USAR. Increases in consequences can only occur when doses to the public are increased beyond those described in Chapter 8 of the USAR. Since stress levels are below allowables, and increases in the stress and strain levels of the bottom spacer disk and support rods do not affect the ability of the transfer cask to maintain its structural integrity, the containment and shielding integrity of the DSC is maintained.

Criticality control has been previously discussed. It has been concluded that criticality control is still maintained over the fuel in the DSC in the event of a cask end drop.

Therefore, the consequences of a cask drop accident previously evaluated in the USAR are not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ YES ☒ NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

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Possibility of New Malfunction: The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity.

The critical functions and cask drop scenarios associated with this change have been discussed previously. No new components, procedures, or tests are being introduced. While the acceleration at which the guide sleeves plastically deform and cease to load the bottom spacer plate increases, all components still perform their functions within the parameters of the licensing basis, that is, with acceptable stress levels computed considering "spacer disk elastic plus plastic deflections" (USAR p. 8.2-12), and analyzing "the vertical top end drop using a bi-linear elastic-plastic model" (SER p.2-48)

Retrievability of fuel has been discussed previously. All spacer disk permanent deflections which could adversely affect the ability of the fuel to be removed from the DSC guide sleeves have been examined and determined to be acceptable.

☐ YES ☒ NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident: The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity.

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in section 8.2 of the USAR, and have been discussed previously. Since there is no change to the design or operation of the NUHOMS system caused by this activity, the possibility of an accident of a different type than any previously evaluated in the SAR will not be created.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ YES ☒ NO Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
2.3	The transfer cask is allowed to be lifted up to 80" in height with a non-single-failure-proof lifting device. The 75 g acceleration envelopes this specification. In the event of a cask drop of 15 inches or more, the DSC is required to be removed from service and inspected, which in turn requires that the fuel be removed from the DSC. Analyses of cask vertical drops performed for this safety evaluation demonstrate that drops of up to 80" can be sustained without unacceptable damage to the cask or DSC, and without decreasing margins of safety. The margin of safety is the difference between the appropriate ASME allowable stress value and the material equivalent failure stress. Table 2.2.3-14 of the SER indicates that the allowable stresses for the DSC components under Combined Accident Loads (Service Level D) are 43.2 ksi (Pri Membrane) and 64.0 ksi (Membrane+Bending). These are provided by ASME for components subject to an elastic analysis. All elastically-computed stresses (Ref. 6) are below these allowable stresses and all margins of safety associated with these allowable stresses remain unchanged. In addition, all plastically-computed stresses (Ref. 6)

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remain below the allowable stresses recommended by ASME for plastically analyzed components. Table 2.2.3-13, Drop Accident Loads (Service Level D), of the SER lists the same allowables (as appear in Table 2.2.3-14) for the spacer disk, but lists the allowable stress for the support rods as 28.0 ksi (Primary). This value, which corresponds to the allowable provided by ASME for primary membrane stresses for certain component supports, is inconsistent with the statement of allowable stresses in BGE calculation C-91-076, Rev. 001, and in Table 2.2.3-14 of the SER. We do not consider this to be an appropriate allowable stress for the support rods because all other technical bases have used component allowable values for the support rods, rather than component support allowable values. (The ASME assigns one set of allowable stresses to components such as vessels, concrete containments, piping, pumps, valves, and storage tanks, and another set to component supports, which are metal elements transmitting component loads to building structures. We believe that the support rods are properly classified as components.) Nonetheless, the computed stress level (Ref. 6) ensures that the margin of safety associated with this allowable stress remains unchanged.

Complete for 72.48:

☐ YES ☒ NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose: A significant increase in occupational dose will not occur as a result of this proposed activity.

The design and operation of the NUHOMS system outside the reactor building is not changed by this proposed activity. Integrity of the DSC or transfer cask is not impacted by this activity. Retrievability of fuel following a design basis 80 inch drop accident is not impacted by this activity. Shielding functions of the DSC and the transfer cask are not impacted by this activity. Because none of these attributes are changed, the occupational doses summarized in USAR Table 7.4-1 are not affected by this activity. Therefore no occupational doses are increased.

☐ YES ☒ NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact: A significant unreviewed environmental impact will not occur as a result of this proposed activity.

Integrity of the DSC or transfer cask is not impacted by this activity. Shielding functions of the DSC and the transfer cask are not impacted by this activity. The proposed activity does not affect any area of the plant site previously undisturbed for the ISFSI, and does not cause any reason for revision to the ISFSI Updated Environmental Report. The proposed activity does not affect the environmental conditions associated with the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: This safety evaluation is prepared to clarify and correct the licensing basis for the NUHOMS-24P system in use at the Calvert Cliffs ISFSI with respect to the postulated transfer cask drop accident.

Proposed changes to the Calvert Cliffs ISFSI USAR include:

- addition of a description of BGE analyses which correct the vertical drop accident analyses in the Topical Report, specifically, of the behavior of the DSC guide sleeves, bottom spacer disk, and support rods.

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- modification of reported stresses in the DSC components resulting from the transfer cask vertical drop.
- updating the reported spacer disk deflection to be expected for a cask drop accident.

Reason for Activity: The transfer cask drop analysis (which is not described in the ISFSI USAR, but is found in the DSC Structural Calculation C-91-076, Rev. 001, and the Topical Report for NUHOMS-24P) assumed, during a vertical cask drop event, that the welds (which attach angle clips from the DSC bottom spacer disk to the DSC guide sleeves) will fail at 35 g's. The independent design review and assessment originally conducted for GPU Nuclear concluded that the approach used to calculate the weld failure point was not conservative, and that the possibility exists that the welds would not fail until a higher acceleration level (> 35g's) is reached. If the welds and angle clips were to remain intact at higher g levels during a vertical cask drop event, then more of the load associated with the deceleration of the DSC guide sleeves would be transferred to the DSC bottom spacer disk and four DSC support rods. As a result, the behavior of these components will change, and the stresses associated with the ISFSI USAR transfer cask drop event will slightly increase.

Activity Summary: Correction of the licensing basis for the NUHOMS-24P system in use at Calvert Cliffs to properly account for the behavior of DSC components, and to correct the stress values that might be expected, in the unlikely event of a cask vertical drop accident, does not constitute an Unreviewed Safety Question (USQ). Increases in stresses and deflections which could occur under these scenarios have been examined and do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. Nor is the possibility for an accident or malfunction of a different type created. The increased stresses do not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification, and, because of the maintenance of DSC structural integrity, criticality control, and retrievability, do not result in any increase in occupational dose. Finally, this activity does not constitute an Unreviewed Environmental Impact.

References:

1. Calvert Cliffs Independent Spent Fuel Storage Installation USAR, Rev. 6
2. SER for the BGE Safety Analysis Report for an ISFSI at Calvert Cliffs, November 1992
3. Topical Report for the Nutech Horizontal Module Storage System for Irradiated Nuclear Fuel NUHOMS-24P, April 1991 (NUH-002, Rev. 1A)
4. BGE Calculation C-91-076, Rev. 001 (Vendor Calc. BGE001.0203, Rev. 004)
5. BGE Calculation C-91-076, Rev. 002 (Vectra Calculation BGE001.0203A, Rev. 0)
6. BGE Calculation CA04132, Rev. 000
7. BGE Issue Report IR1-011-183
8. BGE Drawing 84-001, Rev. 0
9. BGE Drawing 84-019, Rev. 0
10. Reformatted Technical Specifications for Calvert Cliffs ISFSI, May 19, 1992
11. Calvert Cliffs ISFSI Updated Environmental Report, Rev. 1
12. ANSI/ANS-57.9-1992, Design Criteria for an Independent Spent Fuel Storage Installation
13. BGE Memo NEU 98-021, G. E. Gryczkowski to M. J. Gahan, 2/16/98, Subject: Criticality of a Compressed and Flooded Dry Storage Canister (ES199800192-000, Rev. 0000)
14. BGE Memo NFM 98-026, R. H. Beall to M. J. Gahan, 1/23/98, Subject: DSC Guide Sleeve to Fuel Assembly Gap Size
15. BGE IMP I-19, Rev. 6, Spent Fuel Handling Machine Load Weighing System Alignment Test/Adjustment
16. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, 1983.

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ISFSI - Analysis of DSC under Blowdown/Reflood Internal Pressure **72.48 Log No.:SE00136**

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Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions or Bases?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: David A. Scheetz DAScheetz Department: Duke Engineering & Services Date: 2-19-98
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
2-19-98 SIGNATURE / DATE	2/19/98 SIGNATURE / DATE	R.H. Beall 2/19/98 SIGNATURE / DATE

ORA: J.T. Conner 2/19/98

[✓ Approved] Disapproved [✓ Approved] Disapproved

Signature: RK for John Reeves/2-19-98
INDEPENDENT REVIEWER

Signature:
GS-DES, GS-TES, or PE-PDSU

Date 2-19-98

Date 2/20/98

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 98-020 Date: 2-20-98

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 2-20-98
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 2/20/98
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No ✓

Signature: J R Lemons Date: 5/15/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

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ISFSI - Analysis of DSC under Blowdown/Reflood Internal Pressure 72.48 Log No.:SE00136

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Proposed Activity: This safety evaluation is prepared to clarify and correct a non-conforming condition for the NUHOMS system in use at the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) with respect to the Dry Shielded Canister (DSC) internal pressure during blowdown and reflood conditions. The USAR will be changed to revise the pressures and the associated stress values that are expected under blowdown and reflood internal pressures.

Change the Calvert Cliffs ISFSI USAR as follows:

1) USAR Tables 3.6-2 and 3.6-3 will be updated to show that the DSC internal pressures under Service Level B and Service Level D conditions (ref. ASME B&PV Code) include pressures due to blowdown and reflood activities, respectively (40.0 psig for each).
2) USAR Section 8.1.1.1.B will be revised to add a statement that the DSC internal pressure loads include a consideration of pressures due to blowdown and reflood conditions, and that reflood is a Service Level D condition. A re-analysis which includes the effects of these loadings has been conducted, and concludes that these conditions are adequately addressed by the design.
3) USAR Table 8.1-4 will be updated to show the new off-normal stress levels in the DSC components resulting from the blowdown internal pressure load case.
4) USAR Table 8.2-8 will be updated to show the new enveloped stress levels in the DSC components resulting from the blowdown internal pressure.

Reason for Activity:

It was discovered during a calculation review that, while Calvert Cliffs subjects the DSCs to internal pressures of up to 35 psig during blowdown and reflood activities (ref. 12, 13 and 15), the USAR and SER do not mention that these activities cause internal pressure loads. The USAR only describes normal and off-normal internal pressures of 10 psig, and accident internal pressures of 50 psig. This safety evaluation will show, conservatively, that blowdown and reflood pressures up to 40 psig are acceptable.

The blowdown activity occurs after loading fuel into the DSC/transfer cask, moving it to the washdown pit, and welding the shield plug in place. It involves the introduction of filtered, regulated air and/or helium through the vent line into the DSC while borated water discharges through the siphon port. The reflood activity occurs only if the sealed DSC is required to be opened in the washdown pit. It involves the controlled introduction of borated water from the spent fuel pool into the siphon port of the DSC while the monitored helium backfill vents through the DSC vent port.

Blowdown pressure has been determined to be a Service Level B load condition, and reflood pressure a Service Level D load condition (ref. 7 and 11).

These concerns have been documented in BGE Issue Report IR3-005-170.

Function(s) of affected SSC: Summarizing information provided in Chapter 4 of the USAR and ANSI-57.9, the dry shielded canister has containment, shielding, criticality control, configuration control related to fuel retrievability, and thermal safety functions. The primary function of the DSC is to provide containment for the spent nuclear fuel. This is achieved by the stainless steel shell and two inner cover plates (top and bottom ends) which are welded to the shell assembly. There are also outer cover plates (top and bottom) to further assure containment integrity. These are integral with the shield plugs. The DSC provides gamma shielding at its ends by the use of lead shield plugs. These provide ALARA dose rates at the top of the canister during drying and sealing operations and at the bottom for minimizing dose rates during DSC to HSM loading operations and at the HSM door during storage.

Criticality and configuration control is provided by the DSC's internal basket assembly. A series of nine spacer discs and four axial support rods maintain the fuel assemblies in known positions under all normal and accident conditions. The thickness and location of the spacer discs, plus the relative location of the fuel assemblies and the DSC basket material achieve the criticality and configuration control functions. The DSC maintains the helium cover gas which is required for heat rejection and corrosion control. Heat is transferred via thermal radiation and conduction from the fuel through the spacer discs and cover gas to the DSC shell, where it is convectively cooled during HSM interim storage. The presence of helium and spacer discs achieves the heat transfer function.

The on-site transfer cask provides shielding during the DSC closure and transfer operations. The transfer cask also provides structural protection for the DSC against natural and operation hazards during the transfer operations and loading of the DSC into the HSM for interim storage.

ISFSI USAR Revision No.: 6

ISFSI USAR Sections Reviewed: The main chapters reviewed were 3, 4, 5, 7, and 8. The key sections reviewed were 3.6, 4.2, 4.7, 5.1, 7.4, 7.6, 8.1, and 8.2.

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ YES ☒ NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity.

Functions that must be maintained are containment, shielding, criticality control, configuration control related to fuel retrievability, and thermal safety functions. The internal pressurization scenarios impose limiting stresses on some DSC components. Stresses beyond code allowables or unacceptable deflections of safety related DSC equipment/materials define the malfunctions needing to be considered when evaluating this activity involving the internal pressures. The activity in no way degrades the reliability of components important to safety since the design of the DSC is unchanged by this activity. There is no change to the design or operation of the NUHOMS system caused by this activity. While there are increases in stress levels of components of the DSC (ref. 6), the stresses are still below allowable values, and all functions are capable of being performed. Hence, there is no increase in the probability of malfunctions resulting from the accident.

☐ YES ☒ NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity.

The potential malfunctions that could result from this change are the excessive stresses and permanent deformation of the DSC shell and cover plates which could adversely affect canister containment capability. Any breach in this containment capability could have direct radiological consequences, in the form of increased dose rates. The analysis conservatively envelopes the use of both stainless and carbon steel materials for the spacer disks.

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Blowdown pressure has been determined to be a Service Level B load condition, and reflood pressure a Service Level D load condition (ref. 7 and 11). Since the existing design includes a 50 psig accident pressurization, the reflood condition is enveloped by the existing Service Level D analysis. Analysis has shown that DSC shell stresses associated with the blowdown and reflood pressures will increase, but that all stresses are below allowable values (Ref. 6). The stress levels indicate that there is no effect on the containment or structural integrity of the DSC. Hence, the radiological consequences of any component malfunction associated with the blowdown and reflood activities are not increased.

Criticality control is assured as part of the existing design basis and technical specifications by the physical properties and history of the fuel, by mechanical control of the assemblies' locations in the DSC basket, by neutron absorption of the materials of the basket, by Calvert Cliffs administrative controls over fuel identification and handling, and by the presence of soluble boron in the fuel pool for wet operations (Ref. 1). The maintenance of continued positive control over criticality has been examined as part of this evaluation, and has been determined not to be at risk because there is no permanent deformation of any component which could cause changes in any of the criticality control attributes listed above.

The Technical Specification requires retrievability of fuel from the DSC for any drop greater than 15". Since the cask drop is a Service Level D load case and the Service Level D accident pressure envelopes the reflood pressure (refs. 4 and 6), these pressures do not directly affect retrievability. This scenario also envelopes any other scenario for canister unloading.

☐ YES ☒ NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: The probability of occurrence of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity.

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR and in the SER. Of those accidents discussed in the SAR, only HSM air inlet blockage and accidental DSC pressurization scenarios are considered. Since these accidents are caused by wind-blown debris at the HSM and by fuel cladding failure, respectively, their causes are not related to blowdown or reflood activities. There is no change to the design or operation of the NUHOMS system caused by this activity which affects those accidents. Therefore, there is no change in the probability of occurrence of any analyzed accident.

☐ YES ☒ NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident: The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity.

Increases in consequences can only occur when doses to the public are increased beyond those described in Chapter 8 of the USAR. Since no changes will occur to any accidents currently described in the USAR, no increases in the consequences of an accident previously evaluated in the USAR will occur.

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2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ YES ☒ NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction: The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity.

The critical functions associated with this change have been discussed previously. No new components, procedures, or tests are being introduced. While the DSC stresses will increase, they will still be less than allowable values for the appropriate Service Levels, and will not cause the DSC to plastically deform in a manner which inhibits performance of any safety functions. Hence, a malfunction of a different type than any previously evaluated in the USAR will not be created.

☐ YES ☒ NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident: The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity.

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in section 8.2 of the USAR, and have been discussed previously. The only change to the operation of the system as described in the USAR is the inclusion of reflood and blowdown pressures up to 40 psig. As discussed previously, the only effective change is the increase in the pressure used for Service Level B conditions to 40 psig and this change has been fully accounted for in reference 6. Since all resulting stresses are below allowable values, these pressures do not introduce any behavior of the DSC or components which could cause any component to behave in a manner that impedes its functioning. Furthermore, there is no other change to the design or operation of the NUHOMS system caused by this activity, and the possibility of an accident of a different type than any previously evaluated in the SAR will not be created.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ YES ☒ NO Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

The Technical Specifications do not specifically mention the pressures to which the DSC will be exposed during blowdown and reflood conditions. Nonetheless, there is a basis for determining a margin of safety in the design of the DSC components due to design basis pressure loading for ASME Service Levels A, B, C, and D. (Since we earlier identified that only Service Level B pressure loads should change, this discussion only includes Service Level B stresses.) The margin of safety is the difference between the appropriate ASME allowable stress value and the material equivalent failure stress. The specific DSC components which are affected by internal pressures are the DSC shell, the top cover plate, and the bottom cover plate. The SER indicates that the allowable stresses for these components under Service Level B and Combined Service Levels A & B are 18.7 ksi (Primary Membrane),

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28.0 ksi (Membrane + Bending), and 56.1 ksi (Primary + Secondary). These are provided by ASME for components subject to an elastic analysis. All computed stresses (Ref. 6) are below these allowable stresses and all margins of safety associated with these allowable stresses remain unchanged. In addition, Service Level B Primary + Secondary stresses (Ref. 6) for the DSC shell comply with the requirements of ASME III Subsection NB-3653.6, which allows primary plus secondary stresses to exceed $3S_m$ when fatigue is not a factor. The blowdown and reflood loadings occur far too infrequently, for each canister, to make fatigue a factor. Hence, since the allowable stress is not exceeded, the margin of safety remains unchanged.

Complete for 72.48:

☐ YES ☒ NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose: A significant increase in occupational dose will not occur as a result of this proposed activity.

The design and operation of the NUHOMS system is not changed by this proposed activity. Integrity of the DSC or transfer cask is not impacted by this activity. Operational blowdown activities and retrievability of fuel following a drop accident is not impacted by this activity. Shielding functions of the DSC and the transfer cask are not impacted by this activity. Because none of these attributes are changed, the occupational doses summarized in USAR Table 7.4-1 are not affected by this activity. Therefore no occupational doses are increased.

☐ YES ☒ NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact: A significant unreviewed environmental impact will not occur as a result of this proposed activity.

Integrity of the DSC or transfer cask is not impacted by this activity. Shielding functions of the DSC and the transfer cask are not impacted by this activity. The proposed activity does not affect any area of the plant site previously undisturbed for the ISFSI, and does not cause any reason for revision to the ISFSI Updated Environmental Report. The proposed activity does not affect the environmental conditions associated with the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: This safety evaluation is prepared to clarify and correct the licensing basis for the NUHOMS-24P system in use at the Calvert Cliffs ISFSI with respect to the DSC internal pressure during blowdown and reflood conditions.

Proposed changes to the Calvert Cliffs ISFSI USAR include:

- addition of statements which state that DSC internal pressure under Service Level B and D load cases include pressures due to blowdown and reflood conditions.
- modification of reported stresses in the DSC components resulting from the blowdown and reflood pressures.

Reason for Activity: It was discovered, during a calculation review, that DSCs may be subject to up to 40 psig internal pressure during blowdown and reflood activities at Calvert Cliffs, but the USAR and SER describe only normal and off-normal pressure loads of 10 psig, and accident loads of 50 psig.

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ISFSI - Analysis of DSC under Blowdown/Reflood Internal Pressure **72.48 Log No.:SE00136**

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Activity Summary: Correction of the non-conforming condition for the NUHOMS-24P system in use at Calvert Cliffs to revise the stress values that might be expected under DSC blowdown and reflood conditions does not constitute an Unreviewed Safety Question (USQ). Increases in stresses which could occur under these scenarios have been examined and do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. Nor is the possibility for an accident or malfunction of a different type created. The increased stresses do not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification, and, because of the maintenance of DSC structural integrity, criticality control, and retrievability, do not result in any increase in occupational dose. Finally, this activity does not constitute an Unreviewed Environmental Impact.

References:

1. Calvert Cliffs Independent Spent Fuel Storage Installation USAR, Rev. 6
2. SER for the BGE Safety Analysis Report for an ISFSI at Calvert Cliffs, November 1992
3. Topical Report for the Nutech Horizontal Module Storage System for Irradiated Nuclear Fuel NUHOMS-24P, April 1991 (NUH-002, Rev. 1A)
4. BGE Calculation C-91-076, Rev. 001 (Vendor Calc. BGE001.0203, Rev. 004)
5. BGE Calculation C-91-076, Rev. 002 (Vectra Calculation BGE001.0203A, Rev. 0)
6. BGE Calculation CA04132, Rev. 000
7. BGE Issue Report IR3-005-170
8. Reformatted Technical Specifications for Calvert Cliffs ISFSI, May 19, 1992
9. Calvert Cliffs ISFSI Updated Environmental Report, Rev. 1
10. ANSI/ANS-57.9-1992, Design Criteria for an Independent Spent Fuel Storage Installation
11. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, 1983.
12. CCNPP Technical Procedure ISFSI-01, ISFSI Loading, Rev. 3
13. CCNPP Technical Procedure ISFSI-02, ISFSI Unloading, Rev. 3
14. Vectra Calculation BGE001.0224, Rev. 2
15. Vectra Calculation BGE001.0225, Rev. 0

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 1 of 4)

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ACTIVITY: Procedure Change **50.59 Log No.:** SE 00283 **72.48 Log No.:** SE 00139

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions or Technical Specification Bases?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Michael Cox M Cox Department: 47 Date: 4-8-98
PRINTED NAME AND SIGNATURE

☐ YES ☒ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: ______ **PRINTED NAME**	Resp. Ind.: ______ **PRINTED NAME**	Resp. Ind.: ______ **PRINTED NAME**
SIGNATURE	**SIGNATURE**	**SIGNATURE**
Work Group: ______	Work Group: ______	Work Group: ______
Date: ______	Date: ______	Date: ______

Approved ☒ Disapproved ☐	Approved ☒ Disapproved ☐
Signature B.C.Rudell **INDEPENDENT REVIEWER**	Signature 44 **GS-DES,GS-TSES, or PE-PDSU**
Date 4/10/98	Date 4/14 98

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 98-049 Date: 4-20-98

Recommend Approval ☒ Recommend Disapproval ☐ Signature ______ Date 4/20 98
POSRC CHAIRMAN

Approved ☑ Disapproved ☐ Signature ______ Date 4/20/98
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ Yes ☒ No

Signature: ______ Date: 7/16/98
OSSRC SES Chairman

If yes, OSSRC Meeting No.______

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 2 of 4)

Page 2 **of** 5

ACTIVITY: Procedure Change **50.59 Log No.:** SE 00283 **72.48 Log No.:** SE 00139

Proposed Activity: The proposed activity relocates the Life Cycle Management Unit from the Technical Services Engineering Section (Nuclear Engineering Dept) to the Nuclear Project Management Dept

Reason for Activity: The purpose of the change is to achieve better organizational alignment and recognize the important project management aspects of Calvert Cliffs License Renewal efforts.

Function(s) of affected SSC: This proposed organizational change has no direct impact on any SSCs at Calvert Cliffs.

SAR Revision No.: 21

SAR Sections Reviewed: UFSAR Chapter 12

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ Yes ☒ No May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: Relocating the Life Cycle Management Unit to the Nuclear Project Management Dept helps achieve organizational alignment and has no direct impact on SSCs. The Life Cycle Management Unit will continue to perform the necessary functions and analyses needed to pursue License Renewal. Since the functions of the (Cont)

☐ Yes ☒ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: Relocating the Life Cycle Management Unit to the Nuclear Project Management Dept helps achieve organizational alignment and has no direct impact on SSCs. The Life Cycle Management Unit will continue to perform the necessary functions and analyses needed to pursue License Renewal. Since the functions of the organization (Cont)

☐ Yes ☒ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: Relocating the Life Cycle Management Unit to the Nuclear Project Management Dept helps achieve organizational alignment and has no direct impact on SSCs. The Life Cycle Management Unit will continue to perform the necessary functions and analyses needed to pursue License Renewal. Since the functions of the organization (Cont)

☐ Yes ☒ No May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident: Relocating the Life Cycle Management Unit to the Nuclear Project Management Dept helps achieve organizational alignment and has no direct impact on SSCs. The Life Cycle Management Unit will continue to perform the necessary functions and analyses needed to pursue License Renewal. Since the functions of the organization (Cont)

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 3 of 4)

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ACTIVITY: Procedure Change **50.59 Log No.:** SE 00283 **72.48 Log No.:** SE 00139

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ Yes ☒ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction: Relocating the Life Cycle Management Unit to the Nuclear Project Management Dept helps achieve organizational alignment and has no direct impact on SSCs. The Life Cycle Management Unit will continue to perform the necessary functions and analyses needed to pursue License Renewal. Since the functions (Cont.)

☐ Yes ☒ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident: Relocating the Life Cycle Management Unit to the Nuclear Project Management Dept helps achieve organizational alignment and has no direct impact on SSCs. The Life Cycle Management Unit will continue to perform the necessary functions and analyses needed to pursue License Renewal. Since the functions of the organization are not changing, and those functions are administrative, there is no (Cont.)

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ Yes ☒ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
	The functions of the Life Cycle Management Unit are not discussed
	in the Tech Specs.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 4 of 4)

Page 4 of 5

ACTIVITY: Procedure Change **50.59 Log No.:** SE 00283 **72.48 Log No.:** SE 00139

Complete for 72.48:

☐ Yes ☒ No Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose: Relocating the Life Cycle Management Unit to another department is an organizational change that has no direct impact on ISFSI operations. The functions associated with Life Cycle Management remain the same.

☐ Yes ☒ No Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact: Relocating the Life Cycle Management Unit to another department is an organizational change that has no direct impact on ISFSI operations. The functions associated with Life Cycle Management remain the same.

Summary: (For NRC Report, provide a brief overview)

The purpose of this Safety Evaluation is to document our conclusion that relocating the Life Cycle Management Unit from the Nuclear Engineering Dept to the Nuclear Project Management Dept will not require the Tech Specs to be changed nor will it introduce a USQ. This Safety Evaluation is written to recognize that the above described organizational change is different than the current UFSAR description of our organization (see chapter 12.1.5)

CONTINUATION SHEET

Page 5 of 5

~~Comments Continued from Attachment No.:~~ ~~Section:~~ 4/6/98

1.a. Life Cycle Management Unit are not being changed, there is no increase in probability of a malfunction of equipment previously evaluated in the SAR.

1.b. are not being changed, there is no increase in consequences of a malfunction of equipment evaluated in the SAR.

1.c. are not being changed, there is no increase in probability of occurrence of an accident previously evaluated in the SAR.

1.d are not being changed, there is no increase in consequences of an accident previously evaluated in the SAR.

2.a. are not being changed, there is no possibility of creating a malfunction of a different type than any previously evaluated in the SAR. The Life Cycle Management Unit will continue to perform specific analyses that help conclude that SSCs described in the SAR can continue to perform their design functions.

2.b possibility of creating an accident of a different type than any previously evaluated in the SAR. The Life Cycle Management Unit will continue to perform specific analyses that help conclude that SSCs described in the SAR can continue to perform their design functions.

Signature:

Date: 4-9-98

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 1 of 4)

Page 1 **of** 4

ACTIVITY: Org Change **50.59 Log No.:** SE00289 **72.48 Log No.:** SE00140

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?

☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions or Technical Specification Bases?

☒ YES ☐ NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?

☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Michael Cox M. Cox Department: 47 Date: 5-5-98

PRINTED NAME AND SIGNATURE

☐ YES ☒ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: ______	Resp. Ind.: ______	Resp. Ind.: ______
PRINTED NAME	**PRINTED NAME**	**PRINTED NAME**
SIGNATURE	**SIGNATURE**	**SIGNATURE**
Work Group: ______	Work Group: ______	Work Group: ______
Date: ______	Date: ______	Date: ______

Approved ☒ Disapproved ☐

Signature J.C. KILPATRICK

INDEPENDENT REVIEWER

Date 5/18/98

Approved ☒ Disapproved ☐

Signature

GS-DES,GS-TSES, or PE-PDSU

Date 5/18/98

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 98-064 Date: 5-27-98

Recommend Approval ☑ Recommend Disapproval ☐ Signature

POSRC CHAIRMAN

Date 5-27-98

Approved ☑ Disapproved ☐ Signature

PLANT GENERAL MANAGER

Date 5/27/98

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? ☐ Yes ☒ No

Signature:

OSSRC SES Chairman

Date: 7/16/98

If yes, OSSRC Meeting No.______

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 2 of 4)

Page 2 **of** 4

ACTIVITY: Org Change **50.59 Log No.:** SE00289 **72.48 Log No.:** SE00140

Proposed Activity: The proposed activity re-establishes the position of Superintendent - Technical Support. This position is not currently recognized in the UFSAR and will be described in Chap 12.1.1.

Reason for Activity: To increase management oversight in Plant Engineering, Radiation Safety and Chemistry.

Function(s) of affected SSC: N/A

SAR Revision No.: 21

SAR Sections Reviewed: UFSAR Chapter 12, and search on various affected org titles.
ISFSI SAR Chapter 9

Complete for 50.59 and 72.48:

1. **The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.**

☐ Yes ☒ No May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: The proposed activity is an organizational change that will increase management oversight in Plant Engineering, Radiation Safety, and Chemistry. The functions performed by these organizations will remain the same. As such, the probability of occurrence of a malfunction of equipment important to safety will not change.

☐ Yes ☒ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: The proposed activity is an organizational change that will increase management oversight in Plant Engineering, Radiation Safety, and Chemistry. Increasing management oversight in these areas will have no direct impact on the consequences of a malfunction of equipment important to safety.

☐ Yes ☒ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: The proposed activity is an organizational change that will increase management oversight in Plant Engineering, Radiation Safety, and Chemistry. The functions performed by these organizations will remain the same. As such, the probability of occurrence of an accident previously evaluated in the SAR will not change.

☐ Yes ☒ No May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident: The proposed activity is an organizational change that will increase management oversight in Plant Engineering, Radiation Safety, and Chemistry. Increasing management oversight in these areas will have no direct impact on the consequences of an accident previously evaluated in the SAR.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 3 of 4)

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ACTIVITY: Org Change **50.59 Log No.:** SE00289 **72.48 Log No.:** SE00140

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ Yes ☒ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction: The proposed activity is an organizational change that will increase management oversight in Plant Engineering, Radiation Safety, and Chemistry. Increasing management oversight in these areas will have no direct impact on the possibility of a malfunction of a different type than any previously evaluated in

☐ Yes ☒ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident: The proposed activity is an organizational change that will increase management oversight in Plant Engineering, Radiation Safety, and Chemistry. Increasing management oversight in these areas will not create the possibility of an accident of a different type than any previously evaluated in the SAR.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ Yes ☒ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
	The position of Superintendent - Technical Support is not
	described or mentioned in the Tech Specs.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 4 of 4)

Page 4 **of** 4

ACTIVITY: Org Change **50.59 Log No.:** SE00289 **72.48 Log No.:** SE00140

Complete for 72.48:

☐ Yes ☒ No Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose: The proposed activity is an organizational change that will increase management oversight in Plant Engineering, Radiation Safety, and Chemistry. Increasing management oversight in these areas will hopefully result in improved performance in Radiation Safety which will not involve a significant increase in occupational dose.

☐ Yes ☒ No Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact: The proposed activity is an organizational change that will increase management oversight in Plant Engineering, Radiation Safety, and Chemistry. Increasing management oversight in these areas will not involve a significant unreviewed environmental impact.

Summary: (For NRC Report, provide a brief overview)

We are making many improvements to our Radiation Safety Program. To increase management oversight and focus in this area, we are creating the position of Superintendent - Technical Support. The following three functional areas will report to the Superintendent - Technical Support through their respective General Supervisors:

o Plant Engineering
o Radiation Safety
o Chemistry

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 1 of 4)

Page 1 of 4

ACTIVITY: ES199801725-000 Rev 0 | **50.59 Log No.:** | **72.48 Log No.: SE 00144**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: WILLIAM F. LANG William F. Lang Department: 42, NED/PDSU Date: 1/28/99
PRINTED NAME AND SIGNATURE
J. R. SPONSEL 2/8/99

☒ YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp Ind.: Gregory K. Barley PRINTED NAME	Resp Ind.: ROBERT BEALL PRINTED NAME	Resp Ind.: PRINTED NAME
SIGNATURE	SIGNATURE	SIGNATURE
Work Group: Radiation/Chemistry Technical Services	Work Group: Nuclear Fuel Management	Work Group:
Date: 1/29/99	Date: 2/2/99	Date

Approved ☒ Disapproved ☐	Approved ☒ Disapproved ☐
Signature J.E. REMENIUK INDEPENDENT REVIEWER	Signature M. G. POLAK GS-DES, GS-TES, or PE-PDSU
Date 2·2·99	Date 2/9/99

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 99-011 Date: 2/17/99

Recommend Approval ☑	Recommend Disapproval ☐	Signature POSRC CHAIRMAN Date 2/17/99
Approved ☑	Disapproved ☐	Signature PLANT GENERAL MANAGER Date 2/17/99

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ YES ☒ NO
Signature: Date: 5/13/99
OSSRC SES Chairman
If yes, OSSRC Meeting No.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 2 of 4)

Page 2 of 4

ACTIVITY: ES199801725-000 Rev 0000 50.59 Log No.: 72.48 Log No.: SE 00144

Proposed Activity: Revise wording in ISFSI USAR Volume I, Appendix A Environmental Report, Response to NRC question ER-11 item a. The item currently reads that sixteen thermoluminescent dosimeters (TLDs) placed around the ISFSI facility will be read monthly. Change the word "monthly" to "at least quarterly".

Reason for Activity: To resolve a conflict which exists between regulatory documents. The ISFSI TLDs are currently read quarterly. This is per requirements in our Off-site Dose Calculation Manual Attachment 14 which state that the TLDs will be analyzed for gamma dose at least quarterly. The ISFSI Updated Environmental Report (UER), sect. 6.2 states that the ongoing environmental monitoring programs will be expanded and serve as the operational monitoring program for the Calvert Cliffs ISFSI. This is captured in the ISFSI Tech Specs under 6.2 : "The licensee shall include the Calvert Cliffs ISFSI in the environmental monitoring for Calvert Cliffs Nuclear Power Plant. An environmental monitoring program is pursuant to 10 CFR 72.44(d)(2)". From U1 & U2 Tech Specs., Appendix A, 5.5.1.a says: " The Off-site Dose Calculation Manual (ODCM)shall contain the methodology and parameters used in the .... conduct of the radiological environmental monitoring program". Based on prior NRC reviews of CCNPP's ODCM and License amendments, analyzing the ISFSI TLDs on a quarterly frequency is consistent with regulatory requirements and guidance. See attachment A which summarizes the progression of regulatory action in this area and supports a quarterly reading and analyses of TLDs.

At the time that the NRC posed question ER-11., the TLDs around the site and those at the ISFSI were being read monthly. Our radiological environmental monitoring program was being administered by an off-site department at the time. They elected to collect data from the TLDs monthly which was in excess of our regulatory commitment at the time. The answer to question ER-11 merely reflected monitoring practices in practice at that time.

Function(s) of affected SSC: 16 TLDs placed around the ISFSI provide direct radiation monitoring which is part of the ISFSI radiological environmental monitoring program. The TLDs record dose exposure continuously for documentation of compliance with regulatory dose limits. These TLDs are not listed as part of the Calvert Cliffs ISFSI Major Systems, Subsystems and Component listing on Table 1. 3-1 in the ISFSI USAR Volume I.

ISFSI SAR Revision No.: 7 SAR Sections Reviewed: Volume I, Sects. 1, 7, 8, and Appendix A Environmental Report Question ER-11 Volume III (UER) Sects. 5, 6, 7	Tech Spec Bases Rev. No: 1 **Tech Spec Bases Reviewed**: ISFSI Tech Specs: 6.0 Admin. Controls U1 & U2 Tech Specs. 5.0 Admin. Controls and License Amendments #100 and 217 for U1, #82 and 194 for U2

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ YES ☒ NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: This activity constitutes a change in administrative controls involving the radiological environmental monitoring program(REMP). It does not involve the operation, either active or passive of any ISFSI evaluated system, subsystem or component and therefore can not increase the probability of occurrence of a malfunction.

ATTACHMENT 3 SAFETY EVALUATION FORM

ACTIVITY: ES199801725-000 REV. 0000 **72.48 Log No. SE 00144** **pg 3 of 4**

☐ YES ☒ NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: Similar reason as that provided for Probability of Malfunction. Also see discussion under Consequences of Accident which follows.

☐ YES ☒ NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: TLDs monitor radiological dose. The only accident evaluated in the SAR involving radiological consequences is an incredible scenario which assumes that a dry shielded canister (DSC) leaks at the same time that all fuel rods in 24 fuel assemblies rupture due to an event of unspecified origin. Fission gases mainly Kr-85 contained in all the fuel rods of the 24 assemblies are released simultaneously to the environment. The ISFSI TLDs have no relation whatsoever with the probability of this accident occurring.

☐ YES ☒ NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident: In the above described accident involving leakage of fission gases to the environment, the resulting calculated doses are 31 mrem and 148 mrem for the maximum controlled area boundary whole body and thyroid doses, respectively *. These accident doses are well with-in the 10 CFR 72.106 limit of 5000 mrem. Whether the ISFSI TLDs are read monthly or quarterly has no bearing on these accident consequences. The TLDs record dose exposure continuously, so the radiological impact to the environment during the event will still be recorded.

* Source of dose data: NRC Environmental Assessment Report of Calvert Cliff's ISFSI dated 3/22/91

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ YES ☒ NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction: As discussed previously, TLDs are instruments for monitoring radioactive dose and do not interact with any ISFSI systems, subsystems or components described in the SAR.

☐ YES ☒ NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident: ISFSI systems, subsystems or components operate in a passive mode and do not interact with the environmental monitoring TLDs.

ATTACHMENT 3 SAFETY EVALUATION FORM

ACTIVITY: ES199801725-000 REV. 0000 **72.48 Log No. SE 00144** **pg 4 of 4**

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ YES ☒ NO Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
	The portion of the Technical Specifications involved, ISFSI Tech Specs Section 6.2 which discusses the requirement of having ISFSI environmental monitoring involves administrative controls. The bases for these admin controls do not involve margins of safety.

Complete for 72.48:

☐ YES ☒ NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose: This activity does not affect the fuel storage structure shielding nor the manner in which fuel bundles are selected for loading. Therefore inputs into the calculation of the maximally exposed member of the public are unaffected and there can not be an increase in occupational dose. Also the environmental monitoring TLDs are not relied upon to provide direct monitoring for workers involved with loading fuel into the DSCs, transporting the transfer cask or inserting the DSCs into the ISFSI horizontal storage modules. The ISFSI TLDs are used as part of a monitoring program which also includes air samplers, and vegetation and soil sampling to assure public and employee safety during spent fuel storage. The TLDs are considered a continuously monitoring device so it should not matter whether dose data is retrieved monthly or quarterly. An informal test in 1997 compared combined monthly TLD readings over 3 months to quarterly TLD readings at 9 different monitoring locations. The difference in data collected by the 2 methods was small; avg. 6.0% difference.

☐ YES ☒ NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact: Quarterly TLD reading to provide data and support gamma dose analysis for Direct Radiation pathway monitoring complies with CCNPP's ODCM requirements. Refer to page 2 of this evaluation under "Reason for this Activity" on the correlation between the CCNPP ODCM and the ISFSI Environmental Report. This shows there is no significant unreviewed environmental impact.

Summary: (For NRC Report, provide a brief overview)

Safety Evaluation 72.48 Log No. SE 00144
Change wording in ISFSI USAR Volume I, Appendix A Environmental Report, Response to NRC question ER-11 item a. Change the word "monthly" to "at least quarterly" in the following sentence: *TLDs will be read monthly.* This change resolves a conflict between the ISFSI USAR and ISFSI radiological environmental monitoring program requirements in the Calvert Cliffs Offsite Dose Calculation Manual (ODCM). This activity does not involve an unreviewed safety question, nor a significant increase in occupational dose, nor a significant unreviewed environmental impact.

ES199801725-000 REV. 0000

72.48 SE00144

ATTACHMENT A

PROGRESSION OF RADIOLOGICAL ENVIRONMENTAL MONITORING REGULATION AND HOW IT RELATES TO CCNPP's ODCM AND FREQUENCY OF OBTAINING AND ANALYZING DIRECT RADIATION MONITORING DATA

Pre-Licensing Period	10 CFR parts 20 & 50 require environmental monitoring programs be established
DEC 1975	Reg. Guide 4.8, Environmental Tech Specs for Nuclear Power Plants This provided a standard format and principal content of radiological and non-radiological environmental tech specs and surveillances
OCT 1978	NUREG-0133 Preparation of Radiological Effluent Tech Specs for Nuclear Power Plants. This included guidance on preparing the Off-site Dose Calculation Manual(ODCM).
OCT 1978	NUREG-0472 Rev. 1 Draft Radiological Effluent Tech Specs(RETS) for Pressurized Water Reactors Included was a draft Radiological Monitoring Program in table form. Program is based on 26 sampling locations, some on-site, some off-site involving the following exposure pathways and/or sample: Airborne, Direct Radiation, Waterborne, Ingestion. Direct Radiation monitoring involves the use of 2 or more dosimeters or at least 1 instrument continuously measuring and recording dose rate. Monitoring at 8 locations. Sampling, collection and associated gamma dose analysis frequency is at least once per 31 days or at least once per 92 days. Frequency selected is determined by type of dosimeters used.
NOV 1979	The NRC's Radiological Assessment Branch reviewed comments on Reg.Guide 4.8 and issued a Branch Technical Position on the radiological portion of the environmental monitoring program in March 1978. Rev. 1 of the Branch Position paper was issued Nov. 1979 and incorporated lessons learned from the Three Mile Island Accident. This position sets forth an example of an acceptable minimum radiological monitoring program. The most significant difference from the prior NUREGs is that the number of direct radiation monitoring locations expands from 8 to 40. Sampling, collection and analysis frequency is monthly or quarterly. There is a table in this technical position which forms the basis for the format of CCNPP's annual Radiological Environmental Operating Report.
FEB 1985	CCNPP License Amendments #100 (U1) & #82 (U2) are incorporated. Changes were made to the RETS to comply with NRC requirements and intent of NUREG-0133 and NUREG 0472. All requirements of regulation related to RETS were fulfilled. Made reference to our Off-site Dose Calculation Manual (ODCM). The NRC reviewed this and said it was an acceptable reference in that it complied with the methodology and guidelines in NUREG 0133.

ES199801725-000 REV. 0000

72.48 SE00144

ATTACHMENT A

sht 2 of 2

DEC 1989	Initial License Application for the Calvert Cliff's ISFSI submitted. Included in the application is the ISFSI Environmental Report. ISFSI environmental monitoring required by 10 CFR 72.44(d)(2) will be fulfilled by an expansion of the site's Radiological Environmental Monitoring Program(REMP).
JUNE - 1990 NOV	Question and answer period for ISFSI Environmental Report issues. Details on how the site's REMP will be expanded to cover the ISFSI are provided. Use of TLDs is discussed.
MAR 1991	NRC Environmental Assessment Report issued for the ISFSI with finding of no significant impact.
APRIL 1991	NUREG-1301. This issued Generic Letter 89-01, Suppl. 1. Off-site Dose Calculation Manual Guidance: Standard Radiological Effluent Control for Pressurized Water Reactors. In summary this document provides guidance for use by licensees to implement the provision of GL 89-01 which allows them to remove their RETS from the main body of their Tech Specs and place them in the ODCM. It recasted the RETS from the "LCO" format into the "Controls" format of an ODCM entry. Is heavily based on the previous NUREGs and uses the Nov. 1979 Branch technical Position Rev.1 as the basis for a standardized Radiological Environmental Monitoring Program. With respect to Direct Radiation monitoring, sampling, collection and gamma dose analysis frequency is listed as quarterly; with a notation that says: " The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading".
NOV 1992	NRC approves and issues ISFSI License.
OCT 1996	License Amendment #217 (U1) and #194 (U2) issued. These amendments implemented changes to RETS in accordance with Generic Letter 89-01 as described above.

ATTACHMENT 3, SAFETY EVALUATION FORM

NUHOMS-24P DSC Modifications **50.59 Log No.: N/A** **72.48 Log No.: SE00145**
ES199900153 Supplement 001 Revision 000 **Page 1 of 32**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions or Bases?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: RJS for Eric Skowran Department: Sargent & Lundy
Date: 3/26/99
PRINTED NAME AND SIGNATURE

☒ YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	~~Resp. Indv.: C. J. Dobry~~ JTC 3/30/99 ~~Work Group: PES~~	Resp. Indv.: R. H. Beall Work Group: NFM
SIGNATURE / DATE	SIGNATURE / DATE	3/28/99 SIGNATURE / DATE

Approved ☒ Disapproved ☐
Signature: RJ STAKENBORGHS
INDEPENDENT REVIEWER S&L
Date 3/26/99

Approved ☒ Disapproved ☐
Signature: RB. For
GS-DES, GS-TES, or PE-PDSU
Date 3/28/99

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 99-022 Date: 3-31-99

Recommend Approval ☑ Recommend Disapproval ☐ Signature: POSRC CHAIRMAN Date 3-31-99

Approved ☑ Disapproved ☐ Signature: PLANT GENERAL MANAGER Date 4/1/99

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? ☐ Yes ☒ No

Signature: OSSRC SES CHAIRMAN Date: 5/13/99

If yes, OSSRC Meeting No.: ____________

Proposed Activity:

This safety evaluation is prepared to address proposed modifications to the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) Dry Shielded Canister (DSC). The DSC is a major component of the Nutech Horizontal Module System (NUHOMS) spent fuel storage system. Each DSC holds 24 spent fuel assemblies, and provides physical protection and structural support of the spent fuel during loading operations and during storage.

The proposed modifications to the DSC Internal Basket Assembly are being implemented in accordance with Transnuclear West (TNW) Engineering Change Notice (ECN) 98-0516. (The patent for the NUHOMS-24P System design is currently owned by Transnuclear West, Inc.). This activity applies to Calvert Cliffs ISFSI DSCs R025 through R040. The changes that are proposed for the DSC Internal Basket Assembly are intended to ensure that the worst case postulated cask drop accident will not result in deformation of the DSC Internal Basket Assembly to such a degree that post-accident removal of intact fuel assemblies is prohibited. The modifications to the Internal Basket Assembly are identified as follows:

Change No. 1. - Removal of Guide Sleeve Clip Angles
Change No. 2. - Increase the length of the Guide Sleeves
Change No. 3. - Add notched openings to the bottom ends of the Guide Sleeves
Change No. 4. - Add Guide Sleeve Extraction Stops
Change No. 5. - Use continuous (rather than intermittent) longitudinal Guide Sleeve weld seams
Change No. 6. - Increase the dimensional tolerance on the Guide Sleeve envelope
Change No. 7. - Relax the tolerance for the Top Spacer Disk key way dimensions
Change No. 8. - Use a tighter tolerance for the DSC Support Rod diameter

In addition to the DSC Internal Basket Assembly modifications listed above, TNW ECN 98-0516 provides for numerous other changes to the DSC fabrication drawings. The changes are generally categorized as follows:

Change No. 9. - Editorial changes, name changes, and relocation of information within the same document or to another document, and addition of clarifying information without changing intent or technical content
Change No. 10. - Revise DSC component and assembly tolerances, and add true minimum thickness dimensions
Change No. 11. - Revise DSC welding details
Change No. 12. - Provide new component information (Siphon Tube material, Thread Tape & Lubricant)
Change No. 13. - Add leak test requirements for the Top Shield Plug assembly weld
Change No. 14. - Revise the Top Shield Plug lifting post detail
Change No. 15. - Revise the aluminum coating specification for coating carbon steel DSC components

This activity does not affect the design length or tolerance of the DSC internal cavity, nor does it permit storage of spent fuel assemblies that exceed current ISFSI Technical specification requirements.

It is noted that the supporting calculations for this activity are based on a fuel assembly weight of 1450 lbs. ISFSI Technical Specification 3.1.1.(7) prohibits storage of spent fuel assemblies that weigh more than 1300 lbs. Use of a 1450 lb. fuel assembly weight is conservative for the purposes of performing engineering analysis on the DSC. The actual weight of the spent fuel assemblies to be stored for this activity will not exceed the existing Technical Specification weight limit. Future storage of spent fuel assemblies weighing over 1300 lbs. will be handled under a separate licensing action, as appropriate.

ATTACHMENT 3, SAFETY EVALUATION FORM

The ISFSI USAR will be updated to describe the modified DSC design starting with DSC R025. All drawing changes per TNW ECN 98-0516 are considered to be changes to the ISFSI SAR. Other changes to the USAR are summarized as follows:

1) USAR Section 4.2.3.2 describes the Dry Shielded Canister. The description will be updated to identify the modified DSC design that will be used starting with DSC R025.
2) USAR Sections 8.1.1.2, 8.1.1.3, and 8.2.5.2, and Tables 8.1-3, 8.1-4, 8.2-6, 8.2-8, 8.2-9 and 8.2-10 will be amended to add a reference to the modified DSC design. Reference to the BGE supporting analyses will also be added, as appropriate.
3) USAR Section 8.2.5.2 specifies the maximum spacer disk deflection that might be expected under a cask drop accident. This section will be updated to identify the design characteristics of the modified DSC that will ensure the capability to retrieve an intact spent fuel storage assembly following a worst case postulated cask drop accident.
4) USAR Sections 1.1, 1.4, and 9.1.1.3 will be updated to identify Transnuclear West Inc. as the owner of the NUHOMS-24P spent fuel storage cask design.
5) USAR Table 1.3-1 will be updated to provide consistent terminology for the DSC Siphon and Vent Ports components.

Reason for Activity:

Change No.'s 1 through 4:

An independent assessment of Transnuclear West (formerly Nutech Engineers, Pacific Nuclear and Vectra Technologies) was commissioned by GPU Nuclear, Inc., in 1997. The assessment included review of the NUHOMS-24P Topical Report design. During this assessment an issue was raised regarding the behavior of the DSC Internal Basket Assembly during a postulated Transfer Cask (TC) drop accident (reference 25).

The NUHOMS-24P Topical Report DSC design was changed to include removal of Guide Sleeve Clip Angles, increase the length of the Guide Sleeves, add of notched openings to the bottom ends of the Guide Sleeves, and add Guide Sleeve Extraction Stops. The changes are being made in response to a concern regarding the Internal Basket Assembly following a design basis cask drop accident. The geometry of the DSC Internal Basket Assembly following the drop accident must not prohibit retrieval of intact fuel assemblies from the DSC. While the original design was shown to be acceptable, there is limited design margin. The Calvert Cliffs ISFSI uses a NUHOMS-24P site-specific license DSC which is based on the Topical Report, and is affected by this issue.

To understand the DSC design concern, the following description of the existing DSC Internal Basket Assembly is given (also refer to USAR Figure 1.3-1): The major DSC Internal Basket Assembly components include twenty-four stainless steel Guide Sleeves (one for each spent fuel assembly), Nine @ 1.5" thick by 65.5" diameter perforated carbon steel Spacer Disks, and Four @ 3" diameter stainless steel Support Rods. The 9 Spacer Disks are spaced out along the length of the DSC at locations that roughly coincide with the spent fuel assembly spacer grids. The Spacer Disks are not structurally attached to the DSC Shell walls or to the Inner Cover Plates. The 24 Guide Sleeves traverse the length of the DSC cavity through openings in the 9 Spacer Disks. The 4 Support Rods are used to maintain the Spacer Disk locations. The Support Rods traverse the length of the DSC cavity through the 9 Spacer Disks, and are structurally welded to the Spacer Disks at the Support Rod penetration locations. (The Support Rods are not structurally attached to either of the Inner Closure Plates. In the existing design, the Guide Sleeves are fastened to the Bottom Support Disk by welded metal clip angle attachments. Therefore, when the DSC is in the vertical orientation the load path from the Guide Sleeves to the Bottom Inner Cover Plate is as follows: The Guide Sleeves bear upon the Bottom Spacer Disk through the clip angle attachments, the Bottom Spacer Disk (and all of the other 8 Spacer Disks) bear upon the 4 Support Rods, and the Support Rods bear upon the Bottom Inner Closure Plate.

Recent analysis of design basis cask drop accidents revealed that the clip angles that connect the Guide Sleeves to the Bottom Spacer Disk will fail in bending, and will push against the walls of the Guide Sleeve causing the Guide Sleeve to deform (that is, dimpling will occur). In some cases this deformation will cause the clearance between the Guide Sleeve and the spent nuclear fuel assembly to be eliminated. The Guide Sleeve deformation will not cause a rupture of the fuel cladding, but it will take additional force to extract the pinched fuel assemblies during recovery operations. This condition, although tolerable, is undesirable (reference 21). After extensive re-analysis and testing of the DSC components for operating and accident conditions, design changes were advanced for the DSC Internal Basket Assembly design that will preclude Guide Sleeve pinching of the spent fuel assembly as a direct result of a cask drop accident.

The modified DSC Internal Basket Assembly involves the removal of the clip angle attachments between the Guide Sleeves and the Bottom Spacer Disk, thereby eliminating the Guide Sleeve dimpling mechanism. Elimination of the clip angle attachments will allow the Guide Sleeves to slide through the Spacer Disk openings until bearing occurs against either the top or bottom DSC Inner Cover Plate. The Guide Sleeves will bear directly on the DSC Bottom Cover Plate when the DSC is in the vertical orientation. (In the existing design, the Guide Sleeves are supported approximately 1/2" above the Bottom Cover Plate by the welded clip angle attachments.) The modified

DSC increases the overall length of the Guide Sleeves to account for the fact that the Guide Sleeves now rest on the bottom of the DSC. The modified DSC then adds a detail to notch out the bottom of each face of each Guide Sleeve to provide an opening that will facilitate DSC draining and drying.

In order to prevent removal of a Guide Sleeve from the basket in the event a spent fuel assembly becomes stuck during removal, the modified DSC adds new "Extraction Stop" attachments to the Guide Sleeves. The Extraction Stops are nominally 0.75" wide, 2.5" long, 0.06" thick bent metal tabs. Two Extraction Stops will be plug welded to each Guide Sleeve. The Extraction Stops will be positioned on the Guide Sleeve so that they will be located between the 1st and 2nd Spacer Disks from the top of the basket when the Guide Sleeve is installed. The Extraction Stops are not engaged when the DSC is loaded and the DSC Cover Plates are installed. A Guide Sleeve must actually travel upward out of the basket approximately 8" before the Extraction Stop will contact the bottom of the 1st Spacer Disk.

Change No. 5:

The independent assessment of the Transnuclear West NUHOMS-24P Topical Report design identified limited margin in Guide Sleeve longitudinal seam welds (reference 22). The Calvert Cliffs ISFSI site specific license DSC is based on the Topical Report, and is affected by this issue. The Guide Sleeve corner weld that is used to fabricate a Guide Sleeve from plate material was revised from an intermittent weld to a continuous weld, and a minimum effective throat is specified now to ensure adequate weld strength.

Change No.'s 6, 7 and 8:

TNW ECN 98-0516 modifies the following tolerances in order to ease and improve fabrication of the DSC:

Change No. 6 - The Guides Sleeves are formed from a piece (or pieces) of sheet metal that is approximately 22 gauge metal thickness and approximately 155" long. The tolerance on the Guide Sleeve width dimension that is specified in the DSC fabrication drawing will be relaxed from 8.70" (+0.03/-0.03) to 8.70" (+0.06/-0.03) (reference 35). This will allow greater flexibility in fabrication of the Guide Sleeves, but will still be compatible with the DSC Spacer Disk opening dimensions.

Change No. 7 - The Top Spacer Disk Key Way dimensional tolerances were originally specified in accordance with the fabrication drawing default tolerances per ASME Y14.5M-1994. The tolerances on the keyway dimensions are being relaxed to permit a larger opening to ease in fabrication. The new tolerance will allow appropriate clearance following coating the Spacer Disk Key Way surfaces with aluminum thermal spray. The change Spacer to the Disk Key Way tolerance is based on TNW Issues and Lessons learned.

Change No. 8 - The DSC Support Rods are nominally 3" in diameter, and are approximately 158.13" long. The DSC Support Rod diameter tolerance will be "tightened" from 3.00" (+0.05/-0.05) to 3.00" (+0.03/-0.00). Assurance that a minimum 3.00" diameter is maintained is based on TNW Issues and Lessons learned.

Change No. 9:

Change category No. 9 includes changes that do not affect the physical design, testing or operation of ISFSI structures, systems or components (SSCs) important to safety. They are not considered to be factual changes that are subject to the 10 CFR 72.48 safety evaluation process. The proposed changes are identified in this safety evaluation for information only:

- Revise the DSC fabrication specification number and the DSC fabrication drawing numbers referenced in the DSC fabrication drawings. The document number changes are meant to distinguish the modified DSC design to be used for DSCs R25 through R040 from all other DSCs that were previously manufactured for the Calvert Cliffs ISFSI.
- Update the USAR and DSC fabrication drawings to identify Transnuclear West Inc. as the new owner of the NUHOMS-24P spent fuel storage cask design.
- Relocate information and delete redundant information on the DSC fabrication drawings to improve presentation, and to be consistent with Transnuclear West Inc. drawing standards.
- Apply consistent nomenclature for DSC component parts (e.g., use "Siphon Port" in favor of "Drain Port").
- Clarify seal weld inspection requirements by explicitly defining the two liquid penetrant examination levels as "root" and "cover."

Change No. 10:

DSC component tolerances are being revised as summarized in Exhibit E of this safety evaluation. DSC fabrication drawing default block tolerances per ANSI Y14.5M otherwise are still applicable. Reasons for the tolerance revisions are as follows:

- Certain DSC component thickness tolerances are being revised to reflect Transnuclear West Inc. tolerances in favor of ASME and ASTM tolerances. (e.g., Lead Plug Side Casing Plate, Siphon and Vent Port Cover Plates, and the DSC Spacer Disks).
- The DSC Shell Plate, Bottom Cover Plate and the Top Shield Plug Inner Cover Plate tolerances are being changed based on structural engineering evaluation (reference 8).
- The Grapple Ring Plate thickness tolerance will be changed to be consistent with the NUHOMS-24P Topical Report DSC design requirements.
- The Top and Bottom Shield Plug assembly component thickness dimensions and tolerances will be clarified to ensure that minimum design shielding requirements will be maintained.

Change No. 11:

Revise DSC welding details for the following reasons:

Weld Location	Change	Reason
Top Cover Plate to DSC Shell weld	– Reduce weld effective throat – Reduce weld prep and relax weld prep V-angle tolerances	– Minimize base metal distortion – Improve ALARA exposure during field welding – Provide required weld throat under extreme joint fit-up conditions – Dimension and tolerance changes were revised to reflect Transnuclear West Inc. tolerances.
– Top Shield Plug to DSC Shell weld – Top Shield Plug to Siphon and Vent Block weld	– Reduce weld effective throat	– Minimize base metal distortion – Improve ALARA exposure during field welding – Provide required weld throat under extreme joint fit-up conditions
Grapple Ring Plate to Lead Casing Plate weld	Provide alternate detail	Fabrication flexibility
Siphon and Vent Block	Add optional non-structural weld	Fabrication flexibility
– Bottom Cover Plate to DSC Shell weld – DSC Shell Longitudinal and Circumferential welds – Top Shield Plug bent plate built-up weld – Top Shield Plug to DSC Shell weld	Provide weld reinforcement and minimum base metal thickness information, as applicable.	TNW Issues and Lessons Learned relative to weld grinding issues

Change No. 12:

New component information is added for the following reasons:

- The DSC Siphon Tube optional material specification is added for fabrication flexibility.
- The use of Anti-Seize Lubricant in the lifting eyebolt holes is to prevent galling of the threads.
- The Pipe Thread Tape QA classification is added to the fabrication drawing for clarity. The Pipe Thread Tape "non-safety related" classification is consistent with the classification of the components that it will be used on: Swagelock Fittings, and Siphon and Vent Block Ports.

Change No. 13:

Leak test requirements for the Top Shield Plug assembly weld are added to the DSC fabrication drawing to ensure integrity of the pressure boundary. The added leak testing requirements are consistent with USAR 5.1.1.3 leak testing requirements (reference 33).

Change No. 14:

The Top Shield Plug lifting post detail was revised to improve fabrication of the Top Shield Plug.

Change No. 15:

The description of the coating method for coating DSC Internal Basket Assembly carbon steel surfaces with aluminum is being changed from "flame sprayed aluminum" to a generic description of "aluminum thermal spray." This change will allow flexibility for the method of application. The coating material (i.e., aluminum), application location and material thickness requirements are not affected by this change.

Function(s) of affected SSCs:

The DSC and the DSC Internal Basket Assembly are classified as safety-related per 10 CFR 50. The safety-related components of these assemblies include the Spacer Disks, Support Rods, Guide Sleeves, Shield Plugs, and End Closure Plates. The functions of the DSC and the various DSC components are described as follows:

Dry Shielded Canister:

The NUHOMS-24P DSC provides physical protection and structural support of the spent fuel during loading operations, transfer operations, and during storage. It is designed to remain intact under all normal, off-normal and accident conditions identified in the ISFSI USAR. The DSC is designed to perform the following critical functions:

1. Confinement - The DSC design provides mechanical confinement of the stored fuel assemblies to prevent the dispersion of particulate or gaseous radionuclides from the fuel, and to maintain a barrier of helium around the fuel. The primary function of the DSC is to provide confinement of the spent nuclear fuel. This is achieved by the stainless steel shell and two inner cover plates (top and bottom ends) which are welded to the shell assembly. There are also outer cover plates (top and bottom) to further assure

containment integrity. These are integral with the shield plugs. The DSC confinement boundary also is designed to retain the helium cover gas inside the DSC in order to mitigate corrosion of the fuel cladding and prevent expansive oxides from forming in the fuel itself during storage.

2. Criticality Control - The DSC design provides for criticality safety during the wet loading operations, DSC drying operations, and interim storage. This is accomplished by a combination of mechanical separation of the fuel assemblies by the DSC basket assembly, by neutron absorption in the steel guide sleeve material, and administrative controls during the fuel selection process.

3. Fuel Support and Configuration Control - The DSC Internal Basket Assembly provides support for the spent fuel assemblies during normal operations. The DSC also provides configuration control related to post accident recovery of spent nuclear fuel. The DSC is designed so that the worst-case postulated accident, a cask drop accident, will not result in deformation of the Internal Basket Assembly or the DSC shell to such a degree that post-accident removal of intact fuel assemblies is prohibited. The DSC and TC together are designed to limit the deceleration loads on the fuel rods so that their integrity is assured in the worst-case drop accident.

4. Shielding - The DSC materials provide gamma radiation shielding. The DSC provides gamma shielding at its ends by the use of lead shield plugs. These provide ALARA dose rates at the top of the canister during drying and sealing operations and at the bottom for minimizing dose rates during DSC to HSM loading operations and at the HSM door during storage.

5. Thermal - Decay heat is removed by thermal radiation and conduction from the DSC to the TC, and by thermal radiation and conduction and convection from the DSC to Horizontal Storage Module (HSM). The DSC maintains the helium cover gas which is required for corrosion control. This cover gas improves the thermal performance of the DSC.

Internal Basket Assembly:

The functions of the Internal Basket Assembly structures, systems and components are as follows:

1. Guide Sleeves – The Guide Sleeves establish the 24 spent fuel assembly storage compartments within the DSC. The tops of the Guide Sleeves are flared to assist Fuel Handling Operators in guiding the spent fuel assemblies into the sleeves. The inherent neutron absorption capability of the stainless steel guide sleeves provides a measure against criticality. The Guide Sleeves in the existing DSC design are suspended approximately ½" above the Bottom Inner Cover Plate by the Guide Sleeve Attachment Clips. This ½" gap allows for DSC blowdown and drying via the DSC Siphon Port. The proposed modified DSC Guide Sleeves will not be supported by Guide Sleeve Attachment Clips. Therefore the modified DSC Guide Sleeves will have notches cut out at the bottom of each face of each Guide Sleeve to provide an opening that will facilitate DSC draining and drying.

2. Spacer Disks – The Spacer Disks work together with the Guide Sleeves to maintain geometric separation of the fuel assemblies. The Spacer Disks support the weight of the Guide Sleeves, Support Rods and the spent nuclear fuel when the DSC is in a horizontal orientation. The Bottom Spacer Disk in the existing DSC design supports the weight of the 24 Guide Sleeves when the DSC is in a vertical orientation. For the modified DSC design, the only load on the Bottom Spacer Disk when the DSC is in a vertical orientation is due to self weight.

3. Spacer Disk Key Way (old) – The purpose of the Spacer Disk Key Way is to maintain the DSC Internal Basket Assembly orientation during DSC movement. The Key Way doesn't support any safety function.
4. Guide Sleeve Clip Angles - The existing design of the DSC Internal Basket Assembly includes welded metal clip angle attachments between the Guide Sleeves and is attached to the Bottom Spacer Disk. The clip angles were originally designed as a fabrication convenience to restrain the Guide Sleeves during unloaded DSC transport, and during loading and unloading operations (fuel assembly insertion / extraction). The Guide Sleeve Clip Angle attachments would also counteract withdrawal forces in the event of a stuck fuel assembly. The proposed modified DSC Internal Basket Assembly eliminates the use of Guide Sleeve Clip Angle attachments.
5. Support Rods – The Support Rods maintain the Spacer Disk location along the length of the DSC. In the existing DSC design the Support Rods carry the weight of the Guide Sleeves, the Guide Sleeve Clip Angle attachments, and the Spacer Disks when the DSC is in a vertical orientation. In the proposed modified DSC, the Support Rods carry only the weight of the Spacer Disks.
6. Guide Sleeve Extraction Stops (New) – The Guide Sleeves in the modified DSC will have two Extraction Stops (metal tabs) that are mounted to the outside walls of each Guide Sleeve. The extraction stops are intended to prevent Guide Sleeve withdrawal in the event any incidental binding should occur during withdrawal of a spent fuel assembly.

Top Shield Plug Lifting Post:

The Top Shield Plug Lifting Post detail is the interface point between the DSC Top Shield Plug and the Auxiliary Building Spent Fuel Cask Handling Crane. The Top Shield Plug weighs approximately 6,320 lbs. (USAR Table 8.1-1), and therefore, is defined as a "heavy load" per NUREG-0612 (reference 28). During normal DSC fuel loading operations the Top Shield Plug is lifted over spent fuel assemblies in the Transfer Cask while the Transfer Cask is in the spent fuel pool. The Top Shield Plug Lifting Posts are designed to meet the specific heavy load handling design requirements per the Calvert Cliffs Nuclear Power Plant facility operating licenses and associated Technical Specifications pursuant to ISFSI License No. SNM-2505, License Condition No. 18.

ATTACHMENT 3, SAFETY EVALUATION FORM

NUHOMS-24P DSC Modifications **50.59 Log No.: N/A** **72.48 Log No.: SE00145**
ES199900153 Supplement 001 Revision 000

SAR Revision No.: 7

SAR Sections Reviewed:

The main chapters reviewed were 3, 4, 5, 7, and 8. The key sections reviewed are listed as follows:

3.3.4.1	Control Methods for Prevention of Criticality
4.2.1.2	Dry Shielded Canister Structural Specifications
4.2.3.2	Dry Shielded Canister Description
5.1.1.9	Removal of Fuel from the Dry Shielded Canister
8.1.1.2	Dry Shielded Canister Analysis
8.1.1.3	Dry Shielded Canister Internal Basket Analysis
8.2.3.2	Accident Analysis
8.2.5	Cask Drop
8.2.12	Load Combinations
Table 1.3-1	Major Systems, Subsystems and Components of the Calvert Cliffs ISFSI
Table 3.6-3	Summary of Design Criteria for Accident Conditions
Table 8.1-3	Maximum Dry Shielded Canister Stresses for Normal Loads
Table 8.1-4	Maximum Dry Shielded Canister Stresses for Off-Normal Loads
Table 8.2-1	NUHOMS-24P Accident Loading Identification
Table 8.2-6	Maximum Dry Shielded Canister Stresses for Drop Accident Loads
Table 8.2-8	Dry Shielded Canister Enveloping Load Combination Results for Normal and Off-Normal Loads
Table 8.2-9	Dry Shielded Canister Enveloping Load Combination Results for Accident Loads, ASME Service Level C
Table 8.2-10	Dry Shielded Canister Enveloping Load Combination Results for Accident Loads, ASME Service Level D
Appendix A	Q:3.0-2, Load Combination and Design Criteria
Appendix A	Q:BGE001.0203 and Computer Run BPLRWZ, Spacer Disk Analysis

The Calvert Cliffs ISFSI design approval was based upon review of specific design drawings. Drawings listed in Section 1.5 of the Calvert Cliffs ISFSI SER (reference 2) are a part of the ISFSI licensing basis. The DSC design drawing series reviewed for this activity include the following:

BGE-02-1002	(formerly 84-003-E)
BGE-02-1003	(formerly 84-004-E and 84-005-E)
BGE-02-1004	(formerly 84-006-E, 84-007-E and 84-009-E)
BGE-02-1007	(formerly 84-006-E, 84-007-E and 84-009-E)

Although not specifically listed in ISFSI SER Section 1.5, the following drawing series was also reviewed:

BGE-02-1006	(formerly 84-019-E and 84-020-E)

Tech Spec Bases Amendment/Rev No.:

Calvert Cliffs Independent Spent Fuel Storage Installation Technical Specifications, Appendix A to Materials License No. SNM-2505, Amendment 1, July 21, 1995 / (Note: No revision number exists for the ISFSI Tech Spec Bases.)

Tech Spec Bases Reviewed:

2.3 Transfer Cask (TC)
3/4.1 Fuel to be Stored at ISFSI
5.0 Design Features

ATTACHMENT 3, SAFETY EVALUATION FORM

Complete for 50.59 and 72.48:

1. **The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.**

☐ YES ☒ NO **May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Probability of Malfunction: The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity.

Critical functions that must be maintained by the DSC are shielding, thermal safety, confinement, criticality control, and configuration control related to fuel retrievabilty. The failure of DSC structures, systems and components (SSCs) important to safety that could inhibit performance of DSC critical functions has been previously evaluated in USAR Chapter 8. Malfunction of DSC SSCs important to safety will be are evaluated relative to the DSC critical functions as follows:

Shielding:

The critical function related to shielding is not impacted by this activity. The DSC Shell and the DSC Internal Basket Assembly are not credited with augmenting the radiation shielding properties of the DSC (USAR Section 3.3.5.2 and Table 3.3-1). The modified DSC design ensures that the minimum design lead thickness will be maintained in the Top and Bottom Shield Plug assemblies. Therefore, there is no increase in the probability of a malfunction of the DSC shielding due to this activity.

Thermal Control:

The critical function related to thermal safety is not affected by this activity because only minor changes are being made to the DSC Shell and the DSC Internal Basket Assembly geometry and mass. Examples of such changes include increase in the +/- tolerance for the DSC Shell thickness, removal of the Guide Sleeve Clip Angles, addition of 1/2" tall notched openings at the bottom ends of the Guide Sleeves, increase in length of the Guide Sleeves by 1/4", etc. The net effect of these changes on the thermal analysis is negligible in terms of the precision of the engineering analyses. Therefore, there is no increase in the probability of a malfunction of the DSC thermal controls due to this activity

Confinement, Criticality Control & Configuration Control:

Critical functions related to confinement, criticality control and configuration control are predicated on the DSC being able to remain intact under all accident conditions identified in Chapter 8 of the USAR with no loss of function. In order to meet this requirement the DSC is designed for the appropriate loading conditions per the design criteria in USAR Section 3.6, and in accordance with USAR Section 4.2.1.2 structural specifications.

ATTACHMENT 3, SAFETY EVALUATION FORM

Hopper and Associates performed structural calculations to address the proposed changes to the DSC and the DSC Internal Basket Assembly SSCs (references 7, 8 and 27). The scope of the analysis was comprehensive, and covered all safety related DSC and DSC Internal Basket Assembly SSCs. The following components are addressed in the calculations:

- DSC Shell
- Top Lead Liner
- Grapple Ring
- Support Rods
- Top Inner Cover Plate
- Bottom Inner Cover Plate
- Welds
- Guide Sleeves
- Top Outer Cover Plate
- Bottom Inner Liner
- Spacer Disks
- Guide Sleeve Extraction Stops

The DSC design was evaluated in conjunction with ASME B&PVC Section III Subsection NB. ASME Service Levels A, B, C and D for normal, off-normal, emergency and accident level loading conditions and load combinations were imposed on the DSC and the DSC Internal Basket Assembly SSCs. The following is a summary of the significant analyses and results:

- The DSC and DSC Internal Basket Assembly SSCs were evaluated for cask drop, seismic, thermal, pressure, and cask handling loading conditions. The modified DSC dimensions and tolerances were addressed in the analysis where appropriate. The structural analytical methods used for the DSC evaluation are either consistent with or are more conservative than the methods identified in the USAR Chapter 8. The design basis cask drop, thermal and pressure accident level events were found to impose the most limiting structural stresses on the DSC and on the DSC Internal Basket Assembly components. All DSC SSCs were qualified considering stresses due to individual loading conditions added in combination according to Table 3.2-5a of the NUHOMS-24P Topical Report (reference 3). The ASME allowable stresses were determined subject to NRC SER imposed temperature conditions (reference 2). All calculated stresses and stress combinations for the DSC and the DSC Internal Basket Assembly components were determined to be within ASME Section III allowable stress limits (see reference 32 for discussion of exception to this statement). The results of the ASME service level load combination stress analyses are tabulated in Exhibits A through D to this safety evaluation.

- In the existing DSC design there are 24 Guide Sleeves that are connected to the Bottom Spacer Disk through welded Guide Sleeve Clip Angles. The Guide Sleeve loading due to the vertical cask drop event imposes severe stresses on the Bottom Spacer Disk. Because of this, the analytical stress evaluation of the Bottom Spacer Disk for the existing DSC design is performed using a bi-linear elastic-plastic finite element model to determine stresses and plastic deflections (USAR 8.2.5.2 and NRC SER 2-47). This sophisticated modeling technique wasn't required for evaluation of the modified DSC design. The modified DSC design uncouples the Guide Sleeves from the Bottom Spacer Disk (i.e., the Guide Sleeve Clip Angle attachments are removed). This significantly reduces the Bottom Spacer Disk stresses due to the vertical cask drop event. Stresses remain in the elastic range, and a linear elastic finite element analysis is performed. The Spacer Disk stress intensities determined in the Hopper and Associates analysis are within than the ASME code allowable values. More importantly, the removal of the Guide Sleeve Clip Angles eliminates the issues related to Guide Sleeve deformations.

- The Support Rods were evaluated for axial and bending stress, and for critical buckling. The design basis vertical cask drop event imposes the most severe loading condition on the Support Rods. Because the modified DSC uncouples the Guide Sleeves from the Bottom Spacer Disk, the Support Rod stresses are significantly reduced, and stresses are well within ASME allowable limits. The modified DSC design provides greater margin to Support Rod buckling than what is provided by the existing design.

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- Evaluation of axial thermal expansion of the DSC Internal Basket Assembly components relative to the DSC Shell and inner and upper plates was performed by Hopper and Associates, and Transnuclear West Inc. (references 7 and 15). A thermal expansion interference check was also performed to ensure that there is adequate clearance between the Guide Sleeve and the Spacer Disk at the location where the Guide Sleeve passes through the Spacer Disk (reference 27). The evaluations conclude that the proposed modifications to the DSC Internal Basket Assembly will not cause ASME allowable stresses to be exceeded due to differential thermal expansion of components, and that the Guide Sleeves will not subject the Spacer Disks to any out of plane loading due to interference friction (reference 33).

- The Guide Sleeve corner weld seam that is used to fabricate a Guide Sleeve from plate material was revised from an intermittent weld to a continuous weld. This weld detail was evaluated for postulated worst case accident loads, and the weld stresses were found to be within ASME Section III Division 1 Subsection NB allowable stresses.

  The design effective throat for the DSC confinement boundary final closure welds has been reduced in order to improve ALARA exposure, and to ensure the minimum required throat can be developed under the extreme joint fit-up conditions. The affected welds include the Top Cover Plate to DSC Shell weld, the Top Shield Plug to DSC Shell weld and the Top Shield Plug to the Siphon and Vent Block weld. These welds were evaluated with respect to the appropriate allowable stresses, and were found to be acceptable (reference 27). One drawing discrepancy was noted and will be corrected (reference 33).

In conclusion, it was found that the proposed changes to the DSC have been fully analyzed in a manner consistent with USAR design criteria, and the results of the analyses were found to comply with the applicable USAR structural specifications and NRC SER acceptance conditions. Therefore, the probability of occurrence of a malfunction of equipment important to safety will not increase.

☐ **YES** ☒ **NO** **May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Consequences of Malfunction: The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity.

Evaluation of the consequences of malfunctions of equipment important to safety focuses on the DSC critical functions of reactivity control, and configuration control as follows:

Reactivity Control:

Criticality safety during wet loading operations is maintained through the geometric separation of the fuel assemblies within the internal basket assembly, the inherent neutron absorption capability of the stainless steel guide sleeves, the proper selection of sufficiently depleted fuel assemblies and taking credit for the soluble boron in the spent fuel pool. Furthermore, the off-normal reactivity analysis is performed in accordance ANSI/ANS-57.2-1983. This standard imposes conservative assumptions in the analysis for an additional level of safety.

Transnuclear West Inc. has evaluated the proposed changes to the Guide Sleeve and DSC Shell dimensional tolerances for their effect on reactivity (reference 15). The change in the DSC Shell minimum thickness will have the effect of replacing a small thickness of the shell inner diameter with water. The change in the Guide Sleeve inside envelope will have the effect of moving the Guide Sleeve envelope farther away from the fuel, thus increasing the thickness of the water between the fuel and the Guide Sleeve, and increasing the total amount of stainless steel material between the fuel assemblies. The effects of these changes were considered with respect to the design parameters, uncertainties and biases for criticality analysis required per USAR Section 3.3.4. Transnuclear West Inc. determined that the changes will have a negligible effect on criticality, and that the USAR criticality limits will not be exceeded (reference 33).

Section 3.3.4 of the USAR indicates that introduction of a moderator would be necessary to cause a criticality concern during cask handling and storage. Integrity of the confinement boundary was discussed in the preceding question, and it was shown that integrity is assured for the modified DSC design for postulated credible worst case accidents. Nevertheless, criticality analyses have been performed to determine the effects of introduction of moderator into the DSC cavity space (reference 14). Conservativisms in the analysis impose certain deformations on Guide Sleeve wall spacing. The analyses demonstrate that criticality control will be maintained under the most severe accident conditions.

Configuration Control:

There is no change in the normal operation of the NUHOMS-24P system caused by this activity. In the case of performing recovery procedures following the postulated worst case handling accident, the modified DSC Internal Basket Assembly has been evaluated to ensure that a workable geometry will exist so that removal of intact fuel assemblies will not be prohibited. Specifically, there are no unacceptable deflections of the safety related Guide Sleeves, Spacer Disks or Support Rods. Retrievability of intact fuel from the DSC is assured, even following the maximum credible accident.

☐ **YES** ☒ **NO** **May the probability of occurrence of an accident previously evaluated in the SAR be increased?**

Probability of Accident: The probability of occurrence of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity.

Credible accident scenarios that are analyzed for the Calvert Cliffs ISFSI are discussed in Section 8 of the USAR.

The proposed changes to the DSC Internal Basket Assembly have no bearing on the frequency, or on the probability of occurrence of design basis external natural events such as tornado, earthquake, or flood. Similarly, the proposed changes have no bearing on the frequency, or on probability of occurrence of design basis external man-induced events that could affect the ISFSI such as fires, or a Liquid Natural Gas (LNG) plant or pipeline spill or explosion.

The proposed changes to the DSC Internal Basket Assembly do not modify the external configuration of the DSC envelope. The interface between the DSC and the HSM during ISFSI operations and interim storage of the DSC remains unaffected. Therefore, the probability of occurrence of an accident involving loss of HSM air outlet shielding, or blockage of HSM air inlets and outlets will not increase.

Pressurization of the DSC due to fuel cladding failure is an accident scenario identified in USAR Section 8.2.9. The limiting DSC pressurization accident event is a rupture of fuel cladding together with blockage of the HSM vents. As stated in the preceding paragraph, the probability of occurrence of an accident involving blockage of HSM air inlets and outlets will not increase due to the DSC design changes. The USAR does not identify any initiating event for the breach of the fuel cladding for the DSC pressurization accident.

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The modified DSC Internal Basket Assembly will not increase the probability of fuel cladding failure. The Internal Basket Assembly modifications are designed to minimize deformation of the Guide Sleeves to increase retrievability margin in the event of a cask drop accident. There is adequate space to allow for thermal expansion of the DSC Internal Basket Assembly components so that the fuel assemblies will not be subject to any undue stress. In addition, the TC and the DSC were originally designed to limit the deceleration loads on the fuel rods so that their integrity is assured in the worst-case drop accident. The DSC Internal Basket Assembly modifications, in particular, removal of the Guide Sleeve Clip Angles, will have the effect of reducing the overall stiffness of the Internal Basket Assembly. This will work to further reduce the deceleration loads on the fuel assemblies. As a result, the probability of fuel cladding failure due to a cask drop accident will not be increased.

DSC confinement boundary leakage is an accident scenario described in USAR Section 8.2.8. The USAR indicates that there are no credible events that would initiate this type of accident. Despite that, the effects of a failure of all of the fuel rod cladding with a concurrent loss of the DSC confinement boundary is analyzed. As stated in the preceding paragraphs, the probability of an accident that would lead to cladding failure is not increased by the DSC Internal Basket Assembly design changes. The elimination of the Guide Sleeve Clip Angle attachments will have the effect of distributing the weight of the Guide Sleeves more uniformly over the Inner Closure Plate when the DSC is in a vertical orientation thereby reducing the stress concentration on the confinement boundary imposed by the four Support Rods. The confinement boundary plates, structural welds and seal welds have been evaluated in accordance with ISFSI USAR design criteria and structural specifications, and were found to be within ASME code and NRC SER stress limits. Therefore, the modified DSC will not increase probability of DSC confinement boundary leakage.

A cask handling accident involving a drop of the TC is described in USAR Section 8.2.5. There are no heavy load handling system attachment points for the TC that are affected by the DSC modifications. Therefore, the probability of occurrence of a cask drop will not increase due to this activity.

In conclusion, the proposed DSC modifications will not increase the probability of occurrence of any analyzed accident.

☐ **YES** ☒ **NO** **May the consequences of an accident previously evaluated in the SAR be increased?**

Consequences of Accident: The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity.

The relevant ISFSI design basis accidents that are considered for this activity are the cask drop accident event, and the design basis earthquake event.

Cask Drop Accident:

The USAR reports that the TC, DSC, the DSC Internal Basket Assembly and the contained fuel rods will maintain their structural integrity through a cask drop. The USAR conservatively bases the radiological consequences of a cask drop accident on the assumption that the entire TC neutron shield is lost. The change in weight of the modified DSC Internal Basket Assembly will be negligible in terms of the precision of the engineering analysis, and will not contribute to any failure of the TC neutron shielding.

The modified DSC and DSC Internal Basket Assembly have been evaluated to ensure that the worst case postulated cask drop accident will not cause any SSC important to safety to exceed the ASME allowable stresses (subject to NRC SER temperature conditions). The DSC and DSC Internal Basket Assembly have also been evaluated to ensure the design basis cask drop accident will not result in deformation of the DSC, or the DSC Internal Basket Assembly to such a degree that post-accident removal of intact fuel assemblies is prohibited.

The existing DSC Internal Basket Assembly has a condition were Guide Sleeve Clip Angle deformation may cause the Guild Sleeves to "pinch" the fuel assemblies. Although acceptable for certain clip angle sizes, this condition is undesirable. The modified DSC Internal Basket Assembly design will eliminate this potential and increase margin.

The USAR indicates that the structural characteristics of the TC and DSC limit the deceleration loads on the fuel assemblies so that their integrity is assured for the worst case drop accident. The proposed modifications to the DSC Internal Basket Assembly, in particular, removal of the Guide Sleeve Clip Angles, will have the effect of reducing the overall stiffness of the Internal Basket Assembly. This will further reduce the deceleration loads on the stored spent fuel assemblies. Because the geometry of the DSC Internal Basket Assembly will be maintained through the cask drop event, and the structural characteristics of the TC and DSC which limit the deceleration loads on the fuel assemblies will not be negatively affected by the proposed modifications, both criticality control and fuel rod integrity remain assured.

Design Basis Earthquake:

The DSC SSCs important to safety have been designed and evaluated to withstand the forces generated by the design basis earthquake. The analyses use 1.0g vertical seismic acceleration, and 1.5g horizontal seismic acceleration. The forces generated are significantly less than the deceleration forces used in the evaluation of the design basis cask drop event (75g horizontal and 75g vertical deceleration). Hence, the dose consequences due to the design basis earthquake are enveloped by the design basis cask drop accident event.

In conclusion, because the Transfer Cask neutron shielding will remain intact, criticality control is assured, fuel rod integrity is assured, and the retrieval capability of an intact fuel assembly is assured, the on-site and off-site dose consequences will not be increased.

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2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ YES ☒ NO **May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?**

Possibility of New Malfunction: The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity.

The proposed modifications to the DSC and DSC Internal Basket Assembly will add new components, and modify the behavior of existing components under normal operating and accident level conditions. These changes may affect critical functions related to confinement, criticality control and configuration control. These critical functions will be examined to determine if a malfunction of a different type is created.

Confinement:

- Nominal changes in the dimensions of the Guide Sleeves and the Top Spacer Disk Key Way will have no impact on DSC confinement integrity. Thermal interference behavior between the DSC and DSC Internal Basket Assembly components has been checked, and it has been found that ASME allowable stresses will not be exceeded due to differential thermal expansion of components (references 7, 15 and 27). Spacer Disk Key Way stresses have been reviewed and found to be acceptable by Hopper and Associates (reference 7).
- Removal of the Guide Sleeve Clip Angles will allow the Guide Sleeves to rest directly on the Inner Cover Plate when the DSC is in the vertical orientation. With the existing design, the Support Rods support the weight of the Guide Sleeves. The proposed change will reduce punching shear from the Support Rods, and will distribute load on the Inner Closure Plate more uniformly. This change will reduce confinement boundary plate element stresses under normal and accident conditions, and is clearly beneficial.

Criticality Control:

- Removal of the Guide Sleeve Clip Angle attachments will allow the Guide Sleeves to sit flush on the Bottom Inner Cover Plate. Because of this, the modified DSC design adds 1/2" tall notched openings to the bottoms of each Guide Sleeve in order to provide a flow path that will facilitate DSC draining and drying. In the exiting design the Guide Sleeves are suspended approximately 1/2" above the Bottom Cover Plate by the welded clip angle attachments, thus allowing a flow path to permit draining and drying. The difference in flow area between these two configurations will have little impact on the draining and drying times since the controlling parameter is the inner diameter of the Siphon Tube (reference 15). Nevertheless, even if the time to drain the DSC were to increase, there would be no impact on criticality safety. According to USAR Section 5.1.1.3 subcriticality is demonstrated by analysis for all conditions including optimum moderator conditions. (Optimum moderator conditions will be approached if boiling were to occur due to an extended drain down time.)
- USAR Section 3.3.4.1 credits the stainless steel Guide Sleeves with providing some level of criticality control due to their inherent ability to absorb neutrons. Increasing the length of the Guide Sleeves will serve to add negative reactivity to the DSC cavity, and is clearly beneficial.

Configuration Control:

- The proposed modified DSC Internal Basket Assembly removes the Guide Sleeve Clip Angle attachments from the Bottom Spacer Disk. This will allow the Guide Sleeves to sit flush on the Bottom Inner Cover Plate. The modified DSC design adds 1/2" tall notched openings, or scallops, to the bottoms of each Guide Sleeve. Because the bottoms of the Guide Sleeves are scalloped, attention was given to possible buckling failure of the scalloped ends which could result in the Guide Sleeve "pinching" the bottom of a spent fuel assembly. Evaluation of this condition for the most limiting loading condition, the vertical cask drop accident load case, determined that local buckling of the scalloped ends will not occur. The Euler critical buckling load (P_{cr}) for each scalloped section (there are 2 sections at each corner of the Guide Sleeve) was calculated to be 14,000 lbs. This provides a factor of safety greater that 8 against the calculated load of 1,700 lb. for each section under the vertical cask drop loading condition (reference 15 and 27).

  In addition to the Guide Sleeve local buckling concern, overall Guide Sleeve buckling was checked to ensure that column buckling will not "pinch" a fuel assembly. Hopper and Associates calculated that the maximum axial load in the Guide Sleeve due to a vertical cask drop design basis accident event is 12,450 lbs. This load is substantially less than the calculated critical buckling load (P_{cr}) of 48,400 lb. (reference 7, p 209 et. seq.).

- The Guide Sleeve corner weld that is used to fabricate a Guide Sleeve from plate material was revised from an intermittent weld to a continuous weld. This will increase rigidity and improve guide sleeve protection of the spent fuel assembly under normal handling and accident conditions. Because the spent fuel assembly Spacer Grids coincide with the DSC Spacer Disk locations, the additional rigidity will not impact the structural characteristics of the DSC that limit the deceleration loads on the fuel assemblies.

- Two Extraction Stops (metal tabs) are to be plug welded onto each Guide Sleeve. The only intended use for these new components is to prevent removal of a Guide Sleeve from the DSC in the event a spent fuel assembly becomes stuck during withdrawal. Addition of these metal tabs will not alter the behavior of the Guide Sleeves during normal operating and accident conditions. The location of the Extraction Stops has been designed so that they will not cause any additional forces on the Spacer Disks during a vertical cask drop (reference 7).

  Elimination of the clip angles through implementation of the proposed DSC Internal Basket Assembly modifications will ensure that the gap between the Guide Sleeve and the fuel assembly will be maintained following a design basis cask drop accident. Nevertheless, in the event that a fuel assembly becomes stuck, the Guide Sleeve will withdraw along with the fuel assembly for approximately 8" until the Extraction Stop contacts the bottom of the 1st Spacer Disk. Additional extraction force could then be used to free the fuel assembly from the Guide Sleeve.

  Hopper and Associates Engineers performed an extensive evaluation of the effects of the withdrawal forces on the Extraction Stops and on the Guild Sleeves (reference 7). The Extraction Stop welds have been analyzed in accordance with ASME Section III Subsection NF requirements. Extraction loads as high as 800 lbs. will cause only very minor elastic "dimples" in the Guide Sleeves (ANSYS finite element analysis calculated deflection less than or equal to 0.021"). This extraction load cannot be reached as overload protection for the fuel handling machine limits the extraction force on the fuel assembly to 350 lbs. (Ref. 36).

- The Extraction Stops are being added in favor of use of the welded Guide Sleeve Clip Angle attachments. Addition of the Extraction Stops and removal of the Guide Sleeve Clip Angle attachments is clearly beneficial, and the change will not result in a malfunction of a different type.

In conclusion, the addition of new components and changes in the behavior of the existing components will not adversely impact the critical functions of the DSC, and will not result in a malfunction of a different type.

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☐ YES ☒ NO **May the possibility of an accident of a different type than any previously evaluated in the SAR be created?**

Possibility of New Accident: The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity.

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8 of the USAR, and have been discussed previously. Since the operation and performance of the NUHOMS-24P system remain substantially unchanged by this activity, the possibility of an accident of a different type than any previously evaluated in the SAR will not be created.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ YES ☒ NO **Will the margin of safety as defined in the basis for any Technical Specification be reduced?**

Bases: 2.3 and 3/4.1

Discussion of why the margin of safety is not reduced:

NRC Acceptance Limit - Definition:

In order to determine whether there is a reduction in the margin of safety, the NRC acceptance limits that were used in the basis for this Technical Specification must be known. An NRC acceptance limit is the value that is proposed by the licensee in the original SAR, and as modified by the NRC SER. If the NRC SER did not explicitly modify or address the SAR value, then the original SAR value itself is the acceptance limit. The margin may be implicit rather than explicitly expressed as a numerical value. If a specific methodology for computing bounding limits was submitted to the NRC in support of the Calvert Cliffs ISFSI licensing action, reduction in any margin associated with that methodology (i.e., reduction in any specific NRC acceptance condition) would constitute a reduction in the margin of safety (reference 26).

DSC Deceleration - NRC Acceptance Limits:

The method used for predicting the deceleration of a dropped cask is based on EPRI NP-4830 (reference 2 and 30). This method predicts maximum cask deceleration as a function of cask drop height, target hardness, cask orientation and cask weight. The Calvert Cliffs ISFSI bases the maximum deceleration on lifting a loaded Transfer Cask no higher than 80" above a hard concrete surface. The DSC is assumed to be loaded with 24 spent fuel assemblies weighing not more than 1300 lbs. each. Calvert Cliffs ISFSI site specific cask drop analyses were performed to determine the maximum deceleration values for postulated 80" vertical and horizontal drops of the loaded Transfer Cask. USAR Section 8.2.5.2 indicates that the Calvert Cliffs site specific maximum decelerations were determined to be 51 g for the vertical drop case, and 31 g for the horizontal drop case. In spite of these site specific deceleration values, the Calvert Cliffs ISFSI USAR design criteria conservatively uses 75 g for both the vertical and horizontal drop cases, and 25 g for the corner drop case. These conservative deceleration values are based on the NUHOMS-24P Topical Report, and are the basis for NRC SER acceptance of the Calvert Cliffs ISFSI SAR cask drop analyses.(SER 2-46) These deceleration values are provided in USAR Table 3.6-3, Summary of Design Criteria for Accident Conditions.

Therefore, the NRC acceptance limits for cask drop accident deceleration are 75 g horizontal drop, 75 g vertical drop and 25 g corner drop. Use of any lower deceleration values for analysis of Calvert Cliffs ISFSI cask drops would result in a reduction of the margin of safety

It is noted that maximum fuel assembly weight and the Transfer Cask maximum lift height are regulated by the ISFSI Technical Specifications. As such, these parameters will be monitored and complied with as a part of ISFSI fuel loading operations. These Technical Specification limits are not affected by this design activity.

DSC Permissible Stress – NRC Acceptance Limit:

Detailed stress analyses were performed for a series of design basis cask drop accidents for the Calvert Cliffs ISFSI, and were submitted for NRC review and acceptance during initial licensing of the ISFSI facility. The Calvert Cliffs ISFSI SAR employs ASME Section III Division 1 Subsection NB Class 1 Service Level D for determining accident case material allowable stresses for the DSC. (SER 2-49) The NRC acceptance of this methodology was based on use of the worst thermal condition reported in the Topical Report.

Therefore, the NRC acceptance limit for DSC accident case permissible stress is as provided by ASME Section III Division 1 Subsection NB Class 1 Service Level D subject to the specific NRC acceptance condition that the worst thermal conditions (as reported in the Topical Report) must be used in calculating the allowable values. Use of any lower temperatures to compute accident case allowable values, or use of another code or standard that would yield higher allowables for the same conditions would result in a reduction of the margin of safety.

It is noted that the NRC SER imposes similar restrictions on computation of ASME allowable values for emergency conditions,i.e., ASME Service Level C.

DSC Fuel Inspection – NRC Acceptance Limit:

The Calvert Cliffs NUHOMS-24P DSC is designed to meet the requirements of ASME Section III Division 1 Subsection NB Class 1. In the event of a Transfer Cask Drop from a height greater than 15", the fuel must be returned to the spent fuel pool and visually inspected for damage. The basis for this inspection requirement is that permanent deformation of the DSC confinement boundary and DSC internals are permitted under ASME Service Level D loading conditions. Additional basis for this inspection is that the Guide Sleeve Clip Angle welds are predicted to fail at a deceleration of only 35g. The cask drop required fuel inspection height of 15" was deemed to be acceptable based on the due to the robustness of DSC based on the ASME code requirements, and the distance from the postulated worst case cask drop accident drop height. The likelihood that there would be no significant consequence due to a drop from 15" is based on engineering judgement.

Therefore, the NRC acceptance limits for the requirement to inspect fuel following a cask drop of 15 inches is based on compliance with ASME Section III Division 1 Subsection NB Class 1 Service Level D permissible stress limits, and the failure point of DSC Internal Basket Assembly components at 35 g.

Evaluation of the Margin of Safety:

Hopper and Associates performed an extensive re-evaluation of the DSC and the DSC Internal Basket Assembly for the design basis cask drop accident loads (reference 7). All design basis cask drop analyses performed for this activity used deceleration values of 75g for both the horizontal and vertical cask drop cases. The analytical methods used for the evaluation of the effects of those deceleration values on the DSC are either consistent with or are more conservative than the methods identified in the USAR Section 8.2.5. Therefore, the NRC acceptance limits for horizontal, vertical and corner cask drop deceleration are met.

The ISFSI generic and site specific USAR's both maintain that the DSC stresses resulting from a corner drop accident are bounded by the vertical and horizontal drop scenarios. Discussions between BGE and Hopper (reference 31) confirm the rationale why these scenarios bound the corner drop case. The NRC site specific SER lists DSC stress values for the corner drop case which are excerpted from the Transfer Cask load drop case. However, based on the USAR and the Hopper confirmation, BGE has taken the position that it is acceptable to not provide a new DSC Corner Drop analysis.

The scope of the Hopper and Associates structural analyses was comprehensive, and included all safety related confinement boundary and fuel basket assembly components. As stated above, the analytical methods used for the evaluation of cask drops are either consistent with or are more conservative than the methods identified in the USAR Section 8.2.5. Because the proposed modifications to the DSC Internal Basket Assembly uncouple the Guide Sleeves from the Bottom Spacer Disk, additional calculations were performed as required. For example, local and overall buckling of the Guide Sleeve columns was checked as part of the analysis. All calculated stresses for the DSC, and the DSC Internal Basket Assembly components were determined within ASME Section III Division 1 Subsection NB Class 1 Service Level D allowable stress limits when using temperatures for the worst thermal conditions as reported in the Topical Report. The results of the ASME Service Level D load combination stress analyses compared to NRC SER allowable values are tabulated in Exhibit D to this safety evaluation.

The modified DSC Internal Basket Assembly removes the Guide Sleeve Clip Angle attachments. The Clip Angles are considered to be the weakest structural component in the existing DSC design. All safety related Internal Basket Assembly SSCs in the modified design will remain elastic up through the design basis cask drop accidents. Furthermore, as stated in the preceding paragraph, all calculated stresses for the DSC, and the DSC Internal Basket Assembly components were determined to be within ASME Section III Division 1 Subsection NB Class 1 Service Level D allowable stress limits.

In conclusion, because the NRC acceptance limits for the design basis cask drop deceleration values are met, the NRC acceptance limits for DSC material stresses are met, and the NRC acceptance limits identified for fuel inspection are met, the overall margin of safety is not reduced due to this activity.

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Complete for 72.48:

☐ **YES** ☒ **NO** **Will the proposed activity involve a significant increase in occupational dose?**

A significant increase in occupational dose: A significant increase in occupational dose will not occur as a result of this proposed activity.

The operation of the NUHOMS-24P dry cask storage system is not changed by this proposed activity. The proposed changes to the DSC dimensions and tolerances do not reduce the integrity of the confinement boundary, and the DSC radioactive shielding elements (i.e., the shield plugs) are not negatively affected. Changes to the DSC Shell and Guide Sleeve dimensional tolerances will not cause criticality control limits to be exceeded. The DSC changes are designed to ensure Guide Sleeve deflections will remain elastic, and that any deflection due to postulated worst case accident events will not result in any Guide Sleeve pinching a spent fuel assembly. Reduced weld sizes for the Top Cover Plate to DSC Shell weld, the Top Shield Plug to DSC Shell weld, and the Top Shield Plug to Siphon and Vent Block weld will serve to reduce occupational exposure during DSC final closure activities.

In conclusion, the proposed changes to the DSC Internal Basket Assembly do not adversely impact the occupational doses summarized in USAR Table 7.4-1. Therefore, the proposed activity will not significantly increase occupational dose.

☐ **YES** ☒ **NO** **Will the proposed activity involve a significant unreviewed environmental impact?**

A significant unreviewed environmental impact: A significant unreviewed environmental impact will not occur as a result of this proposed activity.

The NUHOMS-24P dry cask storage system confinement and radiological shielding functions remain assured under this proposed activity. The proposed modifications to the DSC and DSC Internal Basket Assembly have been evaluated to ensure confinement boundary stresses will remain within allowable limits under the most severe postulated accident conditions. The changes will ensure that the DSC lead shield plug lead thicknesses will meet design requirements.

The proposed activity does not affect any area of the plant site previously undisturbed for the ISFSI. The proposed activity does not affect any environmental conditions associated with the Calvert Cliffs ISFSI or the Calvert Cliffs Nuclear Power Plant. No new chemicals are being introduced to the ISFSI or the CCNPP as a result of the proposed changes. There are no changes needed for the Calvert Cliffs ISFSI Updated Environmental Report.

Therefore, the proposed activity does not involve an unreviewed environmental impact.

ATTACHMENT 3, SAFETY EVALUATION FORM

References:

1. Calvert Cliffs Independent Spent Fuel Storage Installation USAR, Rev. 7
2. SER for the BGE Safety Analysis Report for an ISFSI at Calvert Cliffs, November 1992
3. Topical Report for the Nutech Horizontal Module Storage System for Irradiated Nuclear Fuel NUHOMS-24P, NUH-002, Revision 1A, July 1989, and NUH-001, Revision 1A, June 1986
4. Calvert Cliffs Independent Spent Fuel Storage Installation Materials License No. SNM-2505, Amendment 1, July 21, 1995
5. Calvert Cliffs Independent Spent Fuel Storage Installation Technical Specifications, Appendix A to Materials License No. SNM-2505, Amendment 1, July 21, 1995
6. Transnuclear West Engineering Change Notice ECN No. 98-0516, Project Title: BGE-02 (DSC), Rev. 0, January 15, 1999
7. Hopper Associates Calculation, Transmittal HABGE-01/99-0745, January 29, 1999
8. Hopper Associates Calculation, Transmittal HABGE-06/98-0655, June 29, 1998
9. BGE Calculation C-91-076, Rev. 001 (Vendor Calc. BGE001.0203, Rev. 004) - Superceded by Ref.11
10. BGE Calculation C-91-076, Rev. 002 (Vectra Calculation BGE001.0203A, Rev. 0) - Superceded by Ref.11
11. BGE Calculation CA04132, Rev. 000 and Rev. 001
12. Calvert Cliffs ISFSI Updated Environmental Report, Rev. 1
13. ANSI/ANS-57.9-1992, Design Criteria for an Independent Spent Fuel Storage Installation
14. BGE Memo NEU 98-021, G. E. Gryczkowski to M. J. Gahan, 2/16/98, Subject: Criticality of a Compressed and Flooded Dry Storage Canister (ES199800192-000, Rev. 0000)
15. Letter from Usama Farradj, Transnuclear Inc., to Robert Beall, BGE, Correction / Clarification of TNW Resolutions of BGE DSC Drawing, Specification and RANOR Procedure Comments, March 16, 1999, BGE02-99-017
16. BGE Memo NFM 98-026, R. H. Beall to M. J. Gahan, 1/23/98, Subject: DSC Guide Sleeve to Fuel Assembly Gap Size
17. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, 1983.
18. Not Used
19. BGE Issue Report IR3-005-206, ISFSI DSC Guide Sleeve Attachments, February 16, 1998
20. BGE Issue Report IR3-007-611, Fuel Assembly Spacer Grids Missalligned with DSC Spacer Disks
21. Hopper Associates Calculation, HABGE-02/98-0610, July 17, 1998
22. Transnuclear Inc. CAR.98.003, Guide Sleeve Undersized Sleeve Weld
23. Not Used
24. BGE Issue Report IR1-011-183
25. Transnuclear Inc. CAR.98.050, DSC Buckling
26. ES-17, Revision 1, 10 CFR 50.59/72.48 Safety Evaluation Screenings/Safety Evaluations (RE: NEI 96-07 p3-6,7,8 &11; NUREG-1606)
27. Letter form K.L. Saunders, Hopper and Associates, to J.T. Conner, BG&E, New DSC Design Review Comments Response, March 22, 1999, HABGE-03/99-0766
28. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, July 1980
29. Transnuclear West Safety Review Screening SR 98-0516 for ECN 98-0516, January 15, 1999
30. Electric Power Research Institute, NP-4830, The Effects of Target Hardness on the Structural Design of Concrete Storage Pads for Spent Fuel Casks, October 1986
31. Telecon, Todd Conner (BGE) to Philip Hasrouni (Hopper & Associates), 3/24/99.
32. Letter from K.L. Saunders, dated March 25, 1999, HABGE-03/99-0767.
33. Letter from Usama Farradj (Transnuclear West) to Robert Beall (BGE), dated March 16, 1999, Resolutions to BGE Comments on DSC Drawings, 9 pages.
34. Letter from Kenneth Saunders (Hopper & Associates) to J. Todd Conner (BGE), dated March 22, 1999, New DSC Design Review Comments Response, 6 pages.
35. Letter from Usama Farradj (Transnuclear West) to Robert Beall (BGE), dated March 23, 1999, Response to BGE Comments on DSC Drawings, 1 page.
36. Setpoint Change Transmittal Sheet, 0-WS-100, Fuel Handling Machine Normal Overload Setpoint Module

ATTACHMENT 3, SAFETY EVALUATION FORM

NUHOMS-24P DSC Modifications	50.59 Log No.: N/A	72.48 Log No.: SE00145
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Exhibit A

DSC Load Combination Results - ASME Service Level A

Stress (ksi)

DSC Component	Stress Type	Hopper and Associates Service Level A Load Case Combinations[a]			ASME Service Level A Allowable[e]
		A2[b]	A3[c]	A4[d]	
DSC Shell	Pri Memb	1.41	1.74	3.53	18.7
	Memb + Bend	9.24	9.67	13.3	28.0
	Memb + Bend+ Thermal	39.2	39.7	43.3	56.1
Bottom Cover Plate	Pri Memb	0.26	1.23	1.53	18.7
	Memb + Bend	1.22	4.78	8.53	28.0
	Memb + Bend+ Thermal	15.2	18.8	22.5	56.1
Top Pressure (Top Cover) Plate	Pri Memb	1.12	1.47	1.13	18.7
	Memb + Bend	1.88	2.26	1.19	28.0
	Memb + Bend+ Thermal	21.8	22.2	21.1	56.1
Top Structural (Top Inner Cover) Plate	Pri Memb	0.19	1.54	2.59	18.7
	Memb + Bend	0.35	2.84	7.31	28.0
	Memb + Bend+ Thermal	22.6	25.0	29.5	56.1
Spacer Disk	Pri Memb	0.57	1.71	1.71	18.7[f]
	Memb + Bend	1.24	3.72	3.72	28.0[f]
	Memb + Bend+ Thermal	33.0	35.5	35.5	56.1[f]
Support Rods	Pri Memb	0.23	0.69	0.69	18.7
	Memb + Bend	--	--	--	28.0
	Memb + Bend+ Thermal	--	--	--	56.1

(a) See Hopper and Associates Calculation, Transmittal HABGE-01/99-0745, January 29, 1999, Table 5.2 – Load Combinations Level A; Load combination A1 is bounded by load combinations A2, A3 and A4.

(b) Combination A2 = DW_2 (DSC with water) + T_{nc} (Inside Cask normal) + P_h (Hydrostatic)

(c) Combination A3 = DW_3 (DSC with fuel) + T_{nc} (Inside Cask normal) + P_a (Normal operating) + L_n (Normal DSC transfer)

(d) Combination A4 = DW_3 (DSC with fuel) + T_{nh} (Inside HSM normal) + P_b (Off –normal blowdown) + L_n (Normal DSC transfer)

(e) See SER for the BGE Safety Analysis Report for an ISFSI at Calvert Cliffs, November 1992, Table 2.2.3-10; Allowable values taken for SA 204 Type 304 material at 400 degrees F.

(f) Allowables are for stainless steel material. Carbon steel material allowables are higher.

Exhibit B

DSC Load Combination Results - ASME Service Level B

Stress (ksi)

		Hopper and Associates Service Level B Load Case Combinations[a]		
DSC Component	**Stress Type**	**B1[b]**	**B2[c]**	**ASME Service Level B Allowable[d]**
DSC Shell	Pri Memb	2.54	4.33	18.7
	Memb + Bend	20.1	23.7	28.0
	Memb + Bend+ Thermal	50.1	53.7	56.1
Bottom Cover Plate	Pri Memb	0.23	0.53	18.7
	Memb + Bend	8.08	11.8	28.0
	Memb + Bend+ Thermal	22.1	25.8	56.1
Top Pressure (Top Cover) Plate	Pri Memb	0.47	0.13	18.7
	Memb + Bend	1.26	0.19	28.0
	Memb + Bend+ Thermal	21.2	20.1	56.1
Top Structural (Top Inner Cover) Plate	Pri Memb	0.54	1.59	18.7
	Memb + Bend	1.84	6.31	28.0
	Memb + Bend+ Thermal	24.0	28.5	56.1
Spacer Disk	Pri Memb	0.57	0.57	18.7[e]
	Memb + Bend	1.24	1.24	28.0[e]
	Memb + Bend+ Thermal	33.0	33.0	56.1[e]
Support Rods	Pri Memb	0.23	0.23	18.7
	Memb + Bend	--	--	28.0
	Memb + Bend+ Thermal	--	--	56.1

(a) See Hopper and Associates Calculation, Transmittal HABGE-01/99-0745, January 29, 1999, Table 5.3 – Load Combinations Level B; Load combinations B3 and B4 are bounded by load combination B1.

(b) Combination B1 = DW_3 (DSC with fuel) + T_{nc} (Inside Cask normal) + P_a (Normal operating) + L_o (Off-normal - Jammed DSC)

(c) Combination B2 = DW_3 (DSC with fuel) + T_{∞} (Inside Cask – off normal) + P_b (Off –normal blowdown) + L_o (Off-normal - Jammed DSC)

(d) See SER for the BGE Safety Analysis Report for an ISFSI at Calvert Cliffs, November 1992, Table 2.2.3-10; Allowable values taken for SA 204 Type 304 material at 400 degrees F.

(e) Allowables are for stainless steel material. Carbon steel material allowables are higher.

Exhibit C

DSC Load Combination Results - ASME Service Level C

Stress (ksi)

		Hopper and Associates Service Level C Load Case Combinations[a]			
DSC Component	Stress Type	C1[b]	C2[c]	C5[d]	ASME Service Level C Allowable[e]
DSC Shell	Pri Memb	4.23	4.23	5.03	21.6
	Memb + Bend	33.8[g]	14.6	25.0	32.4
Top Pressure (Top Cover) Plate	Pri Memb	2.16	2.83	1.83	21.6
	Memb + Bend	6.26	6.54	5.54	32.4
Top Structural (Top Inner Cover) Plate	Pri Memb	2.47	2.99	1.99	21.6
	Memb + Bend	8.9	8.80	7.80	32.4
Bottom Cover Plate	Pri Memb	0.96	1.63	0.63	21.6
	Memb + Bend	7.55	9.78	13.1	32.4
Spacer Disk	Pri Memb	2.00	1.71	0.57	21.6[f]
	Memb + Bend	4.87	3.72	1.24	32.4[f]
Support Rods	Pri Memb	0.69	0.69	0.23	21.6
	Memb + Bend	--	--	--	31.0

(a) See Hopper and Associates Calculation, Transmittal HABGE-01/99-0745, January 29, 1999, Table 5.4 – Load Combinations Level C; Load combinations C3, C4, and C6 are bounded by load combination C5; Secondary stresses (i.e., Q, thermal stresses) are not required for Service Level C.

(b) Combination C1 = DW_3 (DSC with fuel) + T_{nh} (Inside HSM normal) + P_{a1} (Accident – Inner boundary) + E (Seismic)

(c) Combination C2 = DW_3 (DSC with fuel) + T_{nh} (Inside HSM normal) + P_{a1} (Accident – Inner boundary) + L_n (Normal DSC transfer)

(d) Combination C5 = DW_3 (DSC with fuel) + T_{nh} (Inside HSM normal) + P_{a1} (Accident – Inner boundary) + L_o (Off-normal – Jammed DSC)

(e) See SER for the BGE Safety Analysis Report for an ISFSI at Calvert Cliffs, November 1992, Table 2.2.3-12; Allowable values taken at worst case temperatures per the SER, T= 460 degrees F.

(f) Allowables are for stainless steel material. Carbon steel material allowables are higher.

(g) See reference 34 for discussion on the acceptability of this value. Future calculation revision will reduce this value below the SER allowable.

Exhibit D

DSC Load Combination Results - ASME Service Level D

Stress (ksi)

DSC Component	Stress Type	Hopper & Associates D2[(a)]	ASME Service Level D Allowable [(d)]
DSC Shell	Pri Memb Memb + Bend	12.9 63.9	43.2 64.0
Top Structural Plate (Top Inner Cover Plate)	Pri Memb Memb + Bend	14.2 26.5	43.2 64.0
Top Pressure Plate (Top Cover Plate)	Pri Memb Memb + Bend	25.4 63.2	43.2 64.0
Bottom Cover Plate	Pri Memb Memb + Bend	10.7 31.6	43.2 64.0
Spacer Disk	Pri Memb Memb + Bend	41.1 41.1	43.2(e) 64.0(e)
Support Rods	Pri Memb Memb + Bend	17.0 42.0	43.2 64.0
Top End Structural Weld	Primary (Shear)	13.5 (b)	21.6
Bottom End Structural Weld	Primary (Shear)	2.64 (b)	21.6

(a) See Hopper Associates Calculation, Transmittal HABGE-01/99-0745, January 29, 1999, Table 5.5 – Load Combinations Level D; Load Combination D2 bounds Combination D1, therefore D2 values are listed; Combination D2 = DW_3 (DSC with fuel) + T_{nc} (Inside Cask normal) + P_{a2} (Accident – Outer boundary) + DL (Cask Drop)

(b) See Hopper Associates Calculation, Transmittal HABGE-01/99-0745, January 29, 1999, Section 4.6, Weld Analysis

(c) See SER for the BGE Safety Analysis Report for an ISFSI at Calvert Cliffs, November 1992, Table 2.2.3-14; NUTECH and NRC calculated stress values are presented for comparison, and are for information only.

(d) See SER for the BGE Safety Analysis Report for an ISFSI at Calvert Cliffs, November 1992, Table 2.2.3-14; Allowable values taken at worst case temperatures per the SER, T= 460 degrees F.

(e) Allowables are for stainless steel material. Carbon steel material allowables are higher.

Exhibit E

NUHOMS-24P Dry Shielded Canister Component Tolerances

Tolerance (inches)

Component Description	Existing Tolerance	Code Tolerance[b]	TNW ECN 98-0516 Tolerance	Comments
DSC Shell Plate, 5/8" Thk.	+.05/-.05 [a]	+.060/-.010	+.060/-.060	Structural evaluation of the DSC Shell was conservatively based on an overall shell thickness of 0.55" (reference 8). The Shell thickness is not credited for DSC Shielding (USAR Section 3.3.5.2 and Table 3.3-1). Tolerance effect on criticality was evaluated by Transnuclear West Inc. (reference 15).
Bottom Cover Plate, 1¾" Thk.	+.07/-.38	+.075/-.010	+.070/-.125	The tolerance change is enveloped by the existing tolerance. The tolerance change was structurally evaluated (reference 8).
Grapple Ring Plate, 1½" Thk.	+.05/-.01	+.065/-.010	+.05/-.00	The tolerance change is enveloped by the existing tolerance.
Lead Casing Shell Plate, ½" Thk.	+.05/-.05 [a]	+.060/-.010	+.12/-.12	Tolerances change was structurally evaluated (reference 29).
Inner Cover Plate, 1½" Thk.	+.07/-.38	+.075/-.010	+.070/-.125	The tolerance change is enveloped by the existing tolerance. Tolerance change was structurally evaluated (reference 8).
Siphon and Vent Cover Plates, ¼" Thk.	+.05/-.05 [a]	+.050/-.010	+.05/-.01	The tolerance change is enveloped by the existing tolerance.
Lead Plug Side Casing Plate, ½" Thk.	+.05/-.05 [a]	+.060/-.010	+.12/-.01	The minimum tolerance change is more restrictive than the existing tolerance. Tolerances change was structurally evaluated (reference 29).
Spacer Disk Plate, 1½" Thk.	+.05/-.05 [a]	+.080/-.010	+.07/-.01	The minimum tolerance change is more restrictive than the existing tolerance. Tolerances change was structurally evaluated (reference 29).
Support Rod, 3.00" Dia.	+.05/-.05 [a]	+.003/-.003	+.03/-.00	The tolerance change is enveloped by the existing tolerance.

(a) DSC fabrication drawing default block tolerance.
(b) ASTM A480

Summary:

Proposed Activity:

This safety evaluation addresses modifications to the Calvert Cliffs NUHOMS-24P Dry Shielded Canister (DSC). The modifications generally affect the functional design of the DSC Internal Basket Assembly, and affect fabrication details associated with DSC confinement boundary and shield plug components. The design changes to the DSC Internal Basket Assembly include removal of Guide Sleeve Clip Angle welded attachments to the Bottom Spacer Disk, addition of Guide Sleeve Extraction Stops and addition of notched openings to the bottom ends of the Guide Sleeves. There are numerous other changes being made to the DSC fabrication details, including changes to the confinement boundary welding details, changes to various DSC component dimensions and tolerances, designation of consumable materials to be used during DSC fabrication, changes to the shield plug lifting post detail, and identification of certain leak testing requirements. The DSC design documents are also being updated to identify Transnuclear West Inc. as the new patent holder of the NUHOMS-24P system design, and to clarify certain DSC component names. The DSC modifications and document changes are to be implemented beginning with Calvert Cliffs DSC No. R025.

Proposed changes to the Calvert Cliffs ISFSI USAR include the following:

- Addition of a description of the modified DSC Internal Basket Assembly design. The USAR change will specifically describe the elimination of the Guide Sleeve Clip Angle to Bottom Spacer Disk welded attachments, addition of the Guide Sleeve Extraction Stops, and addition of notched openings to the bottom ends of the Guide Sleeve.
- The licensing basis allowable stress criteria for the design of the DSC will be added to the USAR for completeness. The licensing basis allowable stresses are based on ASME Section III Division 1 permissible stresses using the worst thermal conditions reported in the Topical Report.
- Clarify that the existing DSC component stress tables are not applicable for modified DSCs beginning with R025, the tabulated stresses are for information only, and that any calculated DSC stresses must remain within the licensing basis allowable stress criteria.
- Reference to the modified DSC supporting analyses will be added.
- Transnuclear West Inc. will be identified as the new owner of the NUHOMS-24P patent.
- Consistent terminology for the DSC Siphon and Vent Port components will be provided.
- The Calvert Cliffs NUHOMS-24P DSC design drawings will be revised to reflect the modified DSC design.

Reason For Activity:

An independent assessment of the NUHOMS-24P system design determined that the DSC Internal Basket Assembly may not perform as intended during a design basis cask drop accident. Specifically, the clip angle attachments between the Guide Sleeves and the Bottom Spacer Disk could fail in bending and push against the wall of the Guide Sleeves. In some cases the resulting Guide Sleeve deformation could eliminate the clearance between the Guide Sleeve and the spent nuclear fuel assembly. This interference would necessitate the use of additional force to extract the fuel assembly during post accident recovery operations. This condition, although tolerable, is undesirable. The modified DSC Internal Basket Assembly eliminates use of the Guide Sleeve Clip Angle attachments in order to alleviate this type of local Guide Sleeve deformation. Guide Sleeve Extraction Stops are to be added so that in the unlikely event that a fuel assembly does become stuck, the Guide Sleeve will not be withdrawn together with the fuel assembly. The removal of the Guide Sleeve Clip Angles will allow the Guide Sleeves to rest flush on the DSC Bottom Inner Cover Plate. Therefore, the bottoms of the Guide Sleeves will be notched out in order to facilitate DSC draining and drying operations.

Other DSC design changes are deemed to be improvements based an Transnuclear West Inc. design review issues and lessons learned. The changes to the DSC confinement boundary welds are intended to minimize base metal distortion, ease fabrication, and improve ALARA exposure during field welding operations. DSC dimension and tolerance changes are being implemented to reflect Transnuclear West Inc. tolerance standards, to ensure consistency with the Topical Report design, to ensure that minimum design shielding requirements will be maintained, and to fulfill structural engineering evaluation recommendations. The Top Shield Plug Lifting Post detail is being revised to improve fabrication. Incorporation of material and testing information in the DSC fabrication drawings is for clarification of fabrication requirements.

Activity Summary:

The modifications to the Calvert Cliffs NUHOMS-24P DSC Internal Basket Assembly that are intended to alleviate local Guide Sleeve deformation during a design basis cask drop accident, and the various other DSC design and fabrication detail improvements, implemented under this activity, do not result in an Unreviewed Safety Question (USQ). The probability of occurrence of a malfunction of equipment important to safety will not be increased by this activity because the DSC modifications have been fully analyzed in a manner consistent with USAR design criteria, and the results of the analyses were determined to comply with the applicable USAR structural specifications and SER acceptance conditions. The DSC modifications will not increase the probability of occurrence of any analyzed accident. The consequences of an accident will not be increased because radiological shielding is not adversely affected by this modification, criticality control is assured, fuel rod integrity is assured, and the retrieval capability of an intact fuel assembly is assured. The new Guide Sleeve Extraction Stops and the change in the behavior of existing DSC components have been evaluated, and it has been determined that the changes are clearly beneficial, the critical functions of the DSC will not be adversely impacted, and that an accident or malfunction of a different type than any evaluated previously in the SAR is not created by the changes. The DSC modifications do not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification. The DSC modifications do not involve a change in DSC loading operations, and do not adversely impact DSC confinement integrity, shielding features, criticality control, or fuel retrievability, and therefore, will not result in any increase in occupational dose. Finally, this activity does not involve an Unreviewed Environmental Impact.

SAFETY EVALUATION FORM

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ACTIVITY: ES199800308 | **50.59 Log No.: N/A** | **72.48 Log No.: SE00146**

ISFSI – Evaluation of Loaded DSCs With Unanalyzed Clips

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50 59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Jim Remeniuk/ Mohammed Kaiseruddin
PRINTED NAME AND SIGNATURE
Department: CEU/ Sargent & Lundy
Date: 8-25-00

☒ YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye	Resp. Ind.: C. J. Dobry	Resp. Ind.: R H Beall
PRINTED NAME	**PRINTED NAME**	**PRINTED NAME**
SIGNATURE	**SIGNATURE**	**SIGNATURE**
Work Group: Licensing	Work Group: PES	Work Group: NFM
Date: 9/12/00	Date: 9/12/00	Date: 9/12/00

Approved ☒ Disapproved ☐	Approved ☒ Disapproved ☐
Signature	Signature P. D. PATEL
INDEPENDENT REVIEWER	**GS-DES, GS-TSES, or PE-PDSU**
Date 8-23-00 9/13/00	Date 9/14/00

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 00-081 Date: 9/25/00
Recommend Approval ☑ Recommend Disapproval ☐ Signature Date 9/25/00
POSRC CHAIRMAN

Approved ☑ Disapproved ☐ Signature Date 9/26/00
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ YES ☒ NO
Signature: Date: 11/9/00
OSSRC SES Chairman
If yes, OSSRC Meeting No. ____________

Proposed Activity:

The proposed activity consists of making changes to the ISFSI USAR [Ref. 1]. The changes are based on revision to Calculation CA04132 [Ref. 2], which was prepared to evaluate the impact of alternate configurations of clip angle attachments employed in the internal basket assembly of Dry Shielded Canisters (DSCs) R001 through R007, and R010 through R017. (The balance of DSCs R001 through R024 were evaluated in revision 0 of the calculation, and were the subject of SE00133 [Ref. 6]). The subject DSCs are already loaded with fuel assemblies, and are not available for direct measurement of clip angle attachments.

This proposed activity addresses only the DSCs listed above. DSCs R025 and beyond are not affected by this activity due to their new design configuration.

Clip angle attachment arrangements for the subject assemblies are documented in Reference 3, and consist of the following unanalyzed configurations:

- DSCs R001 and R002 utilize tab style clips, in lieu of using clip angles, which were cut from the guide sleeve walls.
- DSCs R003, R004, and R005 have no QA documentation available for the clip angle size measurement.
- DSC R010 utilizes clip angles that are 0.59 inch wide, as opposed to the 0.56 inch that was analyzed previously in revision 0 of Calculation CA04132.
- DSCs R006, R007, and R011 through R017 utilize clip angles with a maximum documented size of 0.56 inch, which was previously analyzed and found to be acceptable.

Appendix L of Reference 2 analyzes the previously unanalyzed configurations. It evaluates the tab style clips of R001 and R002, and a clip angle with a maximum size of 0.625 inch, which is slightly larger than the nominal clip angle size of 0.5 inch plus the maximum tolerance of 0.12 inch [Ref. 4]. This size should envelop clip angle sizes of DSCs R003, R004, R005, and R010, as evident from the tolerances allowed in their design.

The changes being made to the ISFSI USAR consist of the following:

1. Section 8.2.5.2 is being revised to add a note referring to the revised calculation.
2. Tables 8.2-6 and 8.2-10 are being revised to incorporate the new stress values.
3. Section 8.4 is being revised to update Reference 8.23.

Background

The Independent Spent Fuel Storage Installation (ISFSI) at the Calvert Cliffs Nuclear Power Plant (CCNPP) stores DSCs, each of which hold twenty-four old spent fuel assemblies. Each DSC contains an internal basket assembly, which includes twenty-four stainless steel guide sleeves (one for each spent fuel assembly), nine perforated carbon or stainless steel spacer discs, and four stainless steel support rods. The nine spacer discs are spaced out along the length of the DSC at locations that approximately coincide with the spent fuel assembly's eight spacer grids and single lower retention grid. The spacer discs are not structurally attached to the DSC shell walls or inner cover plates. The guide sleeves traverse the length of the DSC cavity through openings in the nine spacer discs. The support rods are used to maintain the spacer disc locations. The support rods traverse the length of the DSC cavity through the nine spacer discs, and are structurally welded to the spacer discs. The guide sleeves are attached only to the bottom spacer disc by four metal clips each, which are welded to the guide sleeves and the bottom side of the bottom spacer disc. These metal clips are the subject of this evaluation. The purpose of the clips is to restrain the guide sleeves, which house the fuel assembly, during DSC transport, and during loading and unloading operations. The clip angle attachments also counteract withdrawal forces in the event of a stuck fuel assembly.

Metal clips for DSCs R008, R009, and R018 through R024 were evaluated earlier in revision 0 of Reference 2, and Reference 6. The evaluation was based on a maximum clip angle size of 0.56-inch (both width and each of the sides). It was determined that, in a vertical top-end drop, the clips would break at a deceleration of 58g (worst case). The clip angles would flatten out before yielding, and would thus pinch the guide sleeves and the fuel assemblies. The fuel assemblies would still be retrievable intact, but would require a pull of 300 lbs. in addition to their weight.

The behavior of clip angles after a vertical drop of the DSC was determined through a test [Ref. 7]. The test verified that the clip angles bend and pinch the guide sleeves and fuel assemblies. The additional pull needed to retrieve a fuel assembly was determined to be 100 lbs. Therefore, the calculated value of the pull force is conservative.

New Analysis

The tab style clips and larger size clip angles analyzed might not break at the previously analyzed deceleration of 58g. Conservatively, it was assumed that they would not break even up to the licensing basis deceleration of 75g. This would exert higher loads on the internal basket assembly components. In addition, the larger size clips would result in a deeper pinching of the fuel assemblies. The behavior of the tab style clips under drop conditions was also analyzed separately.

It was determined for a vertical top-end drop, with the larger size clip angles, that the stresses developed in spacer discs, support rods, and the welds between them would increase but would remain below the allowable values. The fuel assemblies would be pinched deeper, but the fuel rods would not be affected because they are not located in the area where the pinching would occur. The fuel assemblies would remain retrievable intact, with the additional pull needed to be at 297.6 lbs.

(Note: The additional pull needed to retrieve a fuel assembly was determined to be smaller with a larger clip angle size. This is due to the larger cantilever effect that the longer clip angle offered to the force required to bend it back. It would appear then that the conservative assumption on clip angle size would be to make it the smallest necessary to cause pinching. Therefore, the most conservative clip angle size would be equal to the minimum gap between the guide sleeve and the fuel assembly, which is 0.26 inch (1/2 of 0.521) [Ref. 10]. For this clip angle size the additional pull needed to retrieve a fuel assembly is determined by ratioing to be 311.1 lbs.)

The maximum weight of a spent fuel assembly under water is 1188 lbs. [Ref. 15]. Therefore the total maximum force required to pull out the pinched assembly will be (1188+311.1) 1499.1 lbs. Per Reference 13, the Fuel Handling Machine overload setpoint is 1400 lbs., which will not be sufficient to retrieve the heaviest pinched assembly. However, the 1400 lbs. setpoint is based on the maximum safe force that could be applied to fuel assembly spacer grids. In case of a pinched assembly within a guide sleeve, no load will be applied to the spacer grids. Per Reference 16, the limiting force in this case could be as high as 3000 lbs., but should be applied in increments of 500 lbs. as necessary. Therefore, even though the Fuel Handling Machine setpoint may be exceeded, the machine can be cranked manually to retrieve the pinched assembly. ~~(Procedure OI-25A [Ref. 17] needs to be revised to incorporate steps required for this scenario.)~~

MK
9/25/00

9.25.00

The tab type clip angle arrangement was also analyzed for a vertical drop load of 75g. The tab failure mechanism was bulging of the tabs inwards towards the fuel assemblies. The amount of bulging was calculated to be 0.01 inch, which is less than the gap size of 0.26 inch. Therefore, no fuel assembly pinching would occur. The stresses developed in the inner basket assembly components were also calculated to be within the allowables.

Reason for Activity:

This activity is being carried out in part to resolve Issue Report IR3-005-206 [Ref. 5]. The Issue Report was written to address the alternate clip angle arrangements found in DSCs R001 through R007, and R010 through R017, as described above.

The design of the DSC is intended to ensure that the worst case postulated cask drop accident will not result in deformation of the DSC Internal Basket Assembly to such a degree that post-accident retrieval of intact fuel assemblies is not assured or that fuel rod integrity is compromised.

Technical Specifications Section 2.3, "Transfer Cask", states that the transfer cask lifting height, with a non-single-failure-proof lifting device, shall not exceed 80 inches. The Technical Specification also states that for drops greater than 15 inches the DSC will be returned to the spent fuel pool and be visually inspected. Therefore, retrievability of fuel from the DSC needs to be demonstrated.

Function(s) of affected SSC:

The affected systems, structures and components (SSCs) are DSCs R001 through R007 and R010 through R017, and spent fuel assemblies. The DSCs consist of the outer canister and the internal basket assembly, which are classified as important-to-safety per 10 CFR 72. The sub-components of the internal basket assemblies include the Spacer Discs, Support Rods, Guide Sleeves, and clip angle attachments. Changes in the clip angle attachments impact only the internal basket assembly, and not the outer canister of the DSC, because none of the internal basket assembly components are structurally attached to the outer canister.

The DSC provides containment, shielding, criticality control, configuration control related to fuel retrievability, structural support, and thermal safety functions during loading operations, transfer operations, and storage. It is designed to remain intact under all accident conditions identified in the ISFSI USAR with no loss of function. Specific design functions of the DSC include the following:

1. Confinement - The DSC design provides mechanical confinement of the stored fuel assemblies to prevent the dispersion of particulate or gaseous radionuclides from the fuel, and to maintain an inert atmosphere of helium around the fuel. The primary function of the DSC is to provide confinement of the spent nuclear fuel. This is achieved by the stainless steel shell and two inner cover plates (top and bottom ends) which are welded to the shell assembly. There are also outer cover plates (top and bottom) to further assure containment integrity. The DCS confinement boundary also is designed to retain helium cover gas inside the DSC in order to prevent corrosion of the fuel cladding and formation of expansive oxides in the fuel itself during storage. The confinement function is achieved by the outer canister portion of the DSC only, and hence is not affected by changes in the clip angle attachments.
2. Criticality Control - The DSC design provides for sub-criticality during the wet loading, DSC drying, and interim storage operations. This is accomplished by a combination of mechanical separation of the fuel assemblies by the internal basket assembly and neutron absorption in the steel guide sleeve material.

3. Fuel Support and Configuration Control - The DSC internal basket assembly provides support for the spent fuel assemblies during normal operations. The DSC also provides configuration control related to post accident recovery of spent nuclear fuel. The DSC is designed so that the worst-case postulated accidents, including a cask drop, will not result in deformation of the Internal Basket Assembly or the DSC shell to such a degree that retrieval of intact fuel assemblies is not assured. The structural characteristics of the Transfer Cask (TC) and the DSC limit the deceleration loads on the fuel assemblies so that their integrity is assured in the worst-case drop accident.
4. Shielding - The DSC materials provide gamma radiation shielding. The DSC provides gamma shielding at its ends by the use of lead shield plugs. These provide ALARA dose rates at the top of the canister during drying and sealing operations and at the bottom for minimizing dose rates during DSC loading into the Horizontal Storage Module (HSM) and at the HSM door during storage. The shielding function is achieved by the outer canister portion of the DSC only, and hence is not affected by changes in the clip angle attachments.
5. Thermal - Decay heat is removed by thermal radiation and conduction from the DSC to the TC, and by thermal radiation and conduction and convection from the DSC to the HSM. The DSC maintains the helium cover gas, which is required for corrosion control. This cover gas improves the thermal performance of the DSC. The decay heat removal function is achieved by the outer canister portion of the DSC only, and hence is not affected by changes in the clip angle attachments.

Internal Basket Assembly - The functions of the internal basket assembly structures, systems and components are as follows:

1. Guide Sleeves – The guide sleeves establish storage compartments for 24 spent fuel assemblies within the DSC. The tops of the guide sleeves are flared to assist fuel handling operators in guiding the spent fuel assemblies into the sleeves. The guide sleeves are suspended approximately 1/2 inch by the clip angle attachments. This allows for DSC blowdown and drying via the DSC siphon port.
2. Spacer Discs – The spacer discs work together with the guide sleeves to maintain geometric separation of the fuel assemblies. The spacer discs support the weight of the guide sleeves, support rods and the spent nuclear fuel when the DSC is in a horizontal orientation.
3. Clip Angle Attachments (Angles / Tabs) - The DSC internal basket assembly includes metal clip angle attachments or tab style clips, which are welded to the guide sleeves and the bottom spacer disc. The clips are designed as a fabrication convenience to restrain the guide sleeves during unloaded DSC transport, and during loading and unloading operations (fuel assembly insertion / extraction). The clip angle attachments also counteract withdrawal forces in the event of a stuck fuel assembly.
4. Support Rods – The support rods maintain the spacer disk location along the length of the DSC. They carry the weight of the guide sleeves, the clip angle attachments, and the spacer discs when the DSC is in a vertical orientation.
5. Fuel Assembly - The fuel assembly consists of 176 fuel and poison rods, 5 guide tubes, 5 guide tube sleeves, 8 fuel rod spacer grids, upper and lower end fittings, lower retention grid, and a hold-down device. The guide tubes, spacer grids, and end fittings form the structural frame of the assembly. The fuel rod spacer grids maintain the fuel rod pitch over the length of the assembly. The grid provides positive side restraint to the fuel rod but only frictional restraint axially. The spacer grids are the widest part on a fuel assembly. The four outer guide tubes are mechanically attached to the end fittings and the spacer grids are welded to all five guide tubes. The upper end fitting attaches to the guide tubes to serve as an aligning and lifting device for each fuel assembly. The resistance force

is developed by the pinching of the guide sleeve on the lower end fitting. The force to overcome the resistance is transmitted from the upper end fitting through the guide tubes and then to the lower end fitting. No stress is applied to the fuel rods and therefore, pinching of the fuel assembly will not cause rupture of the fuel cladding.

ISFSI USAR Revision No.: 8

ISFSI USAR Sections Reviewed:

The main chapters reviewed were 3, 4, 5, 7, and 8. The key Sections reviewed are listed as follows:

3.3.4.1	Control Methods for Prevention of Criticality
4.2.1.2	Dry Shielded Canister (Structural Specifications)
4.2.3.2	Dry Shielded Canister Description
4.7.3	Individual Unit Description
5.1.1.2	Fuel Loading
5.1.1.9	Removal of Fuel from the Dry Shielded Canister
8.1.1.2	Dry Shielded Canister Analysis
8.1.1.3	Dry Shielded Canister Internal Basket Analysis
8.2.3.2	Accident Analysis
8.2.5	Cask Drop
8.2.12	Load Combinations
Table 3.6-3	Summary of Design Criteria for Accident Conditions
Table 8.1-3	Maximum Dry Shielded Canister Stresses for Normal Loads
Table 8.1-4	Maximum Dry Shielded Canister Stresses for Off-Normal Loads
Table 8.2-1	NUHOMS-24P Accident Loading Identification
Table 8.2-6	Maximum Dry Shielded Canister Stresses for Drop Accident Loads
Table 8.2-8	Dry Shielded Canister Enveloping Load Combination Results for Normal and Off-Normal Loads
Table 8.2-9	Dry Shielded Canister Enveloping Load Combination Results for Accident Loads, ASME Service Level C
Table 8.2-10	Dry Shielded Canister Enveloping Load Combination Results for Accident Loads, ASME Service Level D
Appendix A	Q: 3.0-2, Load Combination and Design Criteria
Appendix A	Q: BGE001.0203 and Computer Run BPLRWZ, Spacer Disk Analysis

Tech Spec Bases Amendment/Rev No.: 2

Calvert Cliffs Independent Spent Fuel Storage Installation Technical Specifications, Appendix A to Materials License No. SNM-2505, Amendment 2, June 30, 2000

Tech Spec Bases Reviewed:

2.3	Transfer Cask (TC)
3/4.1	Fuel to be Stored at ISFSI
5.0	Design Features

SAFETY EVALUATION FORM

ACTIVITY: ES199800308	**50.59 Log No.: N/A**	**72.48 Log No.: SE00146**

ISFSI – Evaluation of Loaded DSCs With Unanalyzed Clips

Complete for 50.59 and 72.48:

1. **The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.**

☐ YES ☒ NO **May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Probability of Malfunction:

The proposed activity consists of evaluating alternate clip angle attachments found in DSCs R001 through R007, and R010 through R017, and revising the stress values applicable to these arrangements. The alternate clip angle attachments consist of clip angles that are larger than those previously analyzed, or tab style attachments that were cut from the guide sleeve walls. The SSCs affected by this activity consist of DSCs and spent fuel assemblies. In the DSCs, the affected sub-components consist of those that belong to the internal basket assembly, namely the guide sleeves, spacer discs, support rods and the clip angle attachment themselves.

The alternate clip angle arrangements were analyzed for drop accident at the licensing basis value of 75g. The larger clip angles caused pinching of the fuel assemblies, and led to larger stresses in the internal basket assembly sub-components. The pinching was found not to impact the fuel rods, because the pinching would occur at the bottom grid, below where the fuel rods are located. Further, the pull force required to retrieve the fuel assemblies was found to be such that the fuel handling machine would be able to handle it within its operating setpoint. The stress values in the spacer discs and the support rods were found to remain within the code allowables. Analysis of neither the tab style clip angle attachments showed that neither pinching of the fuel assemblies occurred nor the stresses increased.

Therefore, the probability of malfunction of equipment important to safety will not be increased.

☐ YES ☒ NO **May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Consequences of Malfunction:

The proposed activity will not lead to breaching of the DSC barrier, or the loss of shielding, so that consequences of malfunction of equipment important to safety, namely the radiation dose to operators or radiation releases from ISFSI will not increase.

USAR Section 3.3.4.1 states that criticality control is assured by the physical properties and history of the fuel, mechanical control of the assembly locations in the DSC basket, neutron absorption of the materials of the basket, Calvert Cliffs administrative controls over fuel identification and handling, and the presence of soluble boron in the fuel pool for wet operations. USAR Section 3.3.4.1 concludes that in the event of a cask end drop, introduction of a moderator would be necessary to cause a criticality concern. A loss of DSC structural integrity would be necessary to allow the introduction of a moderator. Analysis has determined that tabs and nominal and larger size clip angles will not affect DSC structural integrity. Therefore, this activity will not affect criticality control.

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☐ YES ☒ NO **May the probability of occurrence of an accident previously evaluated in the SAR be increased?**

Probability of Accident:

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. They consist of loss of shielding, external missiles, earthquake, flood, cask drop, lightning, blockage of air inlets and outlets, DSC leakage, DSC overpressurization, and forest fire. Of these accidents, only the cask drop accident and earthquake incidents are impacted by this activity. The earthquake scenario is bounded by the cask drop accident, as the acceleration postulated in a design basis earthquake is 1.5g, which is much smaller than the acceleration in the drop accident of 75g. However, the probability of occurrence of cask drop, or any other accident is not increased by the use of alternate clip angle attachments.

There is no change to the design or operation of the NUHOMS system caused by this activity. This activity does not modify the external configuration of the DSC envelope. The interface between the DSC and the HSM during ISFSI operations and interim storage of the DSC remains unaffected. Therefore, the probability of occurrence of an accident involving loss of HSM air outlet shielding, or blockage of HSM air inlets and outlets will not increase.

Pressurization of the DSC due to fuel cladding failure is an accident scenario identified in USAR Section 8.2.9. The limiting DSC pressurization accident event is a rupture of fuel cladding together with blockage of the HSM vents. As stated above, the probability of occurrence of an accident involving blockage of HSM air inlets and outlets will not increase due to this activity. The compression of the fuel assembly occurs at the lower end fitting. The lower retention grid, which houses the fuel rods next to the lower end fitting, is smaller than the lower end fitting. Therefore, the fuel cladding is not impacted and fuel rod integrity is maintained.

DSC leakage is an accident scenario described in USAR Section 8.2.8. The USAR indicates that there are no credible events that would initiate this type of accident. As stated in the preceding paragraphs, the probability of an accident that would lead to cladding failure is not increased by this activity. This activity does not affect the design of the DSC pressure boundary and therefore does not increase the probability of DSC leakage.

☐ YES ☒ NO **May the consequences of an accident previously evaluated in the SAR be increased?**

Consequences of Accident:

The proposed activity, namely analysis and validation of as-built clip angle configurations, will impact the cask drop accident and earthquake incident, as stated above.

The consequences of the cask drop accident with the as-built clip angle configuration were evaluated in Appendix L of Reference 2. The impact on critical safety functions is discussed below. The critical functions affected will be configuration and criticality controls, and confinement. Other critical functions such as shielding and thermal safety are not affected by the clip angle configuration.

Criticality Control: The structural integrity of the fuel assembly was evaluated in Reference 9. The deceleration value used for a drop was 75g, which covers the deceleration imposed on the fuel assembly because of the alternate clip angle configuration. The fuel integrity evaluation is the subject of another safety evaluation, SE00154 [Ref. 18].

Configuration Control: Configuration of fuel assemblies within the DSC needs to be maintained such that the assemblies remain retrievable. The larger size clip angles would bend and pinch the fuel assembly, in the event of a cask top drop accident. It would cause an increase in the extraction force needed to retrieve the fuel assembly. The Fuel Handling Machine is capable of exerting the extra force, although its setpoint would have to be exceeded per the procedure. With the tab style clips, a gap would remain between the guide sleeve and the fuel assembly after a cask drop accident, and hence the fuel assembly retrievability is not affected.

Confinement: The larger size clip angles are anticipated to bend and pinch the fuel assembly at the lower end fitting. The pinching, therefore, will not cause rupture of the fuel cladding. Therefore, the fuel rods would not be damaged, and the radioactive fission products would remain confined.

Shielding: The DSC Internal Basket Assembly is not credited with augmenting the radiation shielding properties of the DSC, and there is no direct interface between the Internal Basket Assembly Components, and the DSC shielding materials. Therefore, there is no impact on the DSC shielding due to this activity.

Thermal Control: The parameters affecting the thermal analysis, such as heat generation or transfer capabilities are not impacted by the proposed activity. Therefore, there is no impact on the DSC thermal control due to this activity.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ YES ☒ NO **May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?**

Possibility of New Malfunction:

The proposed activity examines the malfunction of DSC clip angle attachments following a vertical drop. The malfunction consists of bending of the clips, leading to the pinching of the fuel assemblies as discussed earlier. This malfunction was addressed in Revision 8 of the USAR, and safety evaluation SE00133 [Ref. 6]. The difference in this activity is that the clips are assumed not to break up to 75g, whereas the earlier evaluation established that the clips would fail at about 58g. This difference, however, does not cause a different type of malfunction. The clip angles/tabs bend or deform so as to close the gap between the guide sleeves and the spent fuel assemblies. The clip angles pinch the fuel assemblies at the bottom grid location. Fuel rods are not damaged, however, because they are not located there.

The other impacted SSCs, namely the spacer discs and support rods continue to function as per their design. They continue to provide the structural support to maintain the fuel assembly configuration within the DSC, and keep the fuel assemblies retrievable. The possibility of a malfunction of a different type than any previously evaluated is not created.

☐ YES ☒ NO **May the possibility of an accident of a different type than any previously evaluated in the SAR be created?**

Possibility of New Accident:

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the USAR, and have been discussed previously. Reanalysis of the clips' interaction with the guide sleeve showed slight increases in the stress level in the DSC during a top drop accident. However, since there is no change to the design or operation of the NUHOMS system caused by this activity, the possibility of an accident of a different type than any previously evaluated in the SAR will not be created.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ YES ☒ NO **Will the margin of safety as defined in the basis for any Technical Specification be reduced?**

Tech Spec Bases: 2.3

Discussion of why the margin of safety is not reduced:

Technical Specification Section 2.3, "Transfer Cask", states that the transfer cask lifting height with a non-single-failure-proof lifting device shall not exceed 80 inches, since analyses performed for the DSC and transfer cask confirm that drops of 80 inches could be sustained without unacceptable damage or without decreasing margins of safety. In addition, the technical specification also states that for drops greater than 15 inches the DSC will be returned to the spent fuel pool and be visually inspected, therefore, requiring retrievability of fuel from the DSC for any drop.

Analysis has demonstrated that the clip angles will deform during a top drop accident and cause the guide sleeve to become deformed, reducing the gap between the guide sleeve and the fuel assembly to the point that the guide sleeve pinches the fuel assembly lower end fitting. However, the extraction force calculated to retrieve the fuel assembly is within the capacity of the spent fuel handling equipment, therefore, fuel retrievability can still be assured.

The compression of the guide sleeve on the fuel occurs on the lower end fitting and does not come in contact with the fuel rods, therefore, fuel cladding is unaffected. Stresses to pull the fuel assembly are transmitted through the fuel assembly structure and not the fuel rods.

The results of Calculation CA04132 for the tab style clips showed stresses that are below the allowables. Also, a gap remained between the guide sleeve and the fuel assembly. Therefore, no pinched condition exists between the fuel assembly and the guide sleeve during a top drop accident in DSCs with tab style clips.

The margin of safety is the difference between the appropriate allowable stress value and the material equivalent failure stress. Reference 2 concluded that the computed stress levels are below the allowable stress. The analysis demonstrated that there would be no unacceptable damage to the DSC.

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Therefore, since the fuel assembly can still be retrieved after a top drop accident, computed stresses remain below the allowables, and there is no impact to the integrity of the fuel rods, the margin of safety is not reduced.

Complete for 72.48:

☐ YES ☒ NO **Will the proposed activity involve a significant increase in occupational dose?**

A significant increase in occupational dose:

The radiation protection design and operation of the NUHOMS-24P dry cask storage system is not changed by this proposed activity. The reanalysis of the clips in the DSC Internal Basket Assembly does not reduce the integrity of the confinement boundary, and the radioactive shielding elements (i.e., the shield plugs) are not affected. Retrievability of fuel following a drop is still assured. The larger size clip angles would pinch the fuel assembly, but the fuel rods would not be damaged. Occupational dose associated with post DSC accident recovery is not addressed in the Calvert Cliffs ISFSI USAR. Because none of these attributes are changed, the occupational doses summarized in USAR Table 7.4-1 are not affected by this activity. Therefore, no occupational doses are increased.

☐ YES ☒ NO **Will the proposed activity involve a significant unreviewed environmental impact?**

A significant unreviewed environmental impact:

The NUHOMS-24P dry cask storage system confinement and radiological shielding functions are not reduced by this activity. The DSCs containing the tabs and clip angles, both nominal and larger, have been evaluated to ensure stresses will remain within allowable limits under the most severe postulated drop accident conditions.

This activity does not affect any area of the plant site previously undisturbed for the ISFSI, and does not cause any reason for revision to the ISFSI Updated Environmental Report. This activity does not affect the environmental conditions associated with the ISFSI. Therefore, this activity does not involve an unreviewed environmental impact.

References:

1. Calvert Cliffs Independent Spent Fuel Storage Installation USAR, Rev. 8
2. BGE Calculation CA04132, Rev. 0002, Nutech Horizontal Module System (NUHOMS) 24P ISFSI Dry Shielded Canister Structural Analysis for DSC Assemblies R001-R024
3. Transnuclear Inc. Letter No.113-114-98-018, from Usama Farradj to W. J. Lippold of BGE, 01/29/1998
4. Drawing 84-002-E, NUHOMS-24P ISFSI DSC Basket Assembly, Rev. 0, 11/08/1993
5. BGE Issue Report IR3-005-206, 02/15/1998
6. 10CFR72.48 Safety Evaluation No. SE00133, ISFSI – Analysis of DSC Under Postulated Cask Drop Accident
7. Test Report for the BGE DSC Guidesleeve Standard Clip Design Test, Prepared by J. E. Remeniuk, 02/12/1998
8. Technical Specifications for Calvert Cliffs ISFSI, Amendment 2
9. BGE Calculation CA04678, Rev. 00, BGE (Calvert Cliffs Units 1 & 2) ISFSI Dry Storage Cask Drop Analysis
10. BGE Memo NFM 98-026, R. H. Beall to M. J. Gahan, 1/23/98, Subject: DSC Guide Sleeve to Fuel Assembly Gap Size
11. BGE IMP I-19, Rev. 7, Spent Fuel Handling Machine Load Weighing System Alignment Test/Adjustment
12. UFSAR/USAR/TSB Change Request, UCR No. 00038, for ESP ES199800308-000
13. Setpoint Change Transmittal, 0-WS-100, Fuel Handling Machine Overload Setpoint
14. BGE Drawing 84-019-E, NUOHMS-24P ISFSI DSC Carbon Steel Basket Assembly, Rev. 1, 4/15/1998
15. BGE Memorandum NEU 99-164, from T. A. Schearer to G. V. Patel, et al., 7/7/1999
16. Combustion Engineering Letter from J. E. Baum to J. A. Mihalcik of BG&E, Allowable Lifting Force for a Stuck Fuel Assembly, June 28, 1989
17. Calvert Cliffs Nuclear Power Plant Procedure OI-25A, Spent Fuel Handling Machine, Revision 22, 8/14/00
18. 10CFR72.48 Safety Evaluation No. SE00154, ISFSI – Evaluation of Fuel Assembly Integrity
19. Calvert Cliffs ISFSI Updated Environmental Report, Rev. 1

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ACTIVITY: ES199800308	50.59 Log No.: N/A	72.48 Log No.: SE00146

ISFSI – Evaluation of Loaded DSCs With Unanalyzed Clips

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: The proposed activity consists of making changes to the ISFSI USAR, in order to account for the impact of alternate configurations of clip angle attachments. The alternate configurations were employed in Dry Shielded Canisters (DSCs) R001 through R007, and R010 through R017. These configurations consisted of tab style clips that were cut from the guide sleeve walls (in lieu of clip angles), and clip angles of sizes larger than those previously analyzed.

The changes being made to the ISFSI USAR consist of providing reference to the new analysis, and incorporating its results.

Reason for Activity: This activity is being carried out in part to address the alternate clip angle attachments found in DSCs, as described above in the Proposed Activity.

One of the functions of the clip angle attachment is to break on a vertical top-end cask drop to prevent overstressing of internal basket of the DSCs. The concern with the alternate clip angle attachment arrangements is that the attachments may not break at the deceleration value previously determined, thereby transmitting larger loads to the DSC components. In addition, the larger size clips would result in a deeper pinching of the fuel assemblies, which may affect the fuel assembly retrievability.

Technical Specification Section 2.3, "Transfer Cask", states that the transfer cask lifting height with a non-single-failure-proof lifting device shall not exceed 80 inches. The technical specification also states that for drops greater than 15 inches the DSC will be returned to the spent fuel pool and be visually inspected. Therefore, retrievability of fuel from the DSC following a drop needs to be demonstrated

Activity Summary: A new analysis was performed assuming that the clip angle attachments would not break up to the licensing basis deceleration of 75g. This would exert higher loads on the internal basket assembly components. The behavior of the tab style clips under drop conditions was also analyzed separately.

It was determined for a vertical top-end drop that, with the larger size clip angles the stresses developed in spacer discs, support rods, and the welds between them would increase but would remain below the allowable values. The fuel assemblies would be pinched deeper, but the fuel rods would not be affected because they are not located in the area where the pinching would occur, namely the lower end fitting. The fuel assemblies intact retrievable would require an additional pull force of approximately 300 lbs., which the Fuel Handling Machine is capable of exerting.

The tab type clip arrangement was also analyzed for a vertical top-end drop load of 75g. The tab failure mechanism was bulging of the tabs inwards towards the fuel assemblies. The amount of bulging was calculated to be less than the gap between the guide sleeve and the fuel assembly. Therefore, no fuel assembly pinching would occur. The stresses developed in the inner basket assembly components were also calculated to be within the allowables.

USQ Determination: This activity was evaluated against the criteria of 10CFR72.48(a)(2), such as the probability of occurrence or the consequences of an accident or the malfunction of equipment important to safety, and it was concluded that it does not involve an unreviewed safety question (USQ).

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ACTIVITY: ES199800466	**50.59 Log No.: N/A**	**72.48 Log No.: SE00148**

ISFSI DSC Grid Misalignment

Based on the attached discussion, does this activity

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Jim Rementuk/ Mohammed Kaiseruddin
PRINTED NAME AND SIGNATURE
Department: CEU/ Sargent & Lundy
Date: 7/28/00
8·25·00

☒ YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye	Resp. Ind.: C. J. Dobry	Resp. Ind.: R. H. Beall
PRINTED NAME	**PRINTED NAME**	**PRINTED NAME**
SIGNATURE	**SIGNATURE**	**SIGNATURE**
Work Group: Licensing	Work Group: PES	Work Group: NFM
Date: 9/12/00	Date: 9/13/00	Date: 9/10/00

Approved ☒ Disapproved ☐
Signature 9/13/00
INDEPENDENT REVIEWER
Date ~~8-23-00~~ 9/13/00

Approved ☒ Disapproved ☐
Signature P. D. PATEL
GS-DES, GS-TSES, or PE-PDSU
Date 9/14/00

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.. 00-081 Date: 9/25/00
Recommend Approval ☑ Recommend Disapproval ☐ Signature **POSRC CHAIRMAN** Date 9/25/00

Approved ☑ Disapproved ☐ Signature **PLANT GENERAL MANAGER** Date 9/26/00

The OSSRC has reviewed this evaluation according to NS-2-100
Full OSSRC Committee review required? ☐ YES ☒ NO
Signature: OSSRC SES Chairman Date: 11/9/00
If yes, OSSRC Meeting No.

Proposed Activity:

The proposed activity consists of making changes to the ISFSI USAR [Refs. 1 and 2]. The changes are based on Calculation CA04132 [Ref. 3], which was prepared, in part, to evaluate the misalignment of some of the fuel assembly spacer grids in relation to the spacer discs provided within the Dry Shielded Canisters (DSCs) R001 through R024. The USAR Section 1.3.1.1 states that the DSC spacer disks are "at intervals corresponding to the fuel assembly spacer grids". This was found not to be the case for a few of the spacer grids of fuel assemblies from batches 1/2 D/E/F/G/H, 1J, 2A, 2B and 2C.

Misalignment was found in fuel assembly lower end fittings (LEFs) and the DSC bottom spacer discs; and in fuel assembly eighth zircalloy grids and the DSC eighth spacer discs. The worst case misalignment consisted of

- the mid-plane of the LEF flow plate (load bearing surface) being 0.9742 inch lower than the bottom surface of the bottom spacer disc, and
- the bottom of eighth zircalloy grid being 1.1017 inches lower than the DSC eighth spacer disc.

Section 4.7 of Reference 3 evaluated the consequences of misalignment of grids. Horizontal drop of the transfer cask (TC) was considered a limiting case because of the geometry. Because of misalignment, the most vulnerable DSC component would be the guide sleeve.

The changes being made to the ISFSI UFSAR consist of revisions to Sections 1.3.1.1 and 8.2.5.2 to add statements regarding the misalignment of grids of fuel assemblies and DSCs, and its analysis.

Stress values reported in Tables 8.2-6 and 8.2-10 envelope the stresses from misalignment configuration, therefore no changes are necessary.

Background

The Independent Spent Fuel Storage Installation (ISFSI) at the Calvert Cliffs Nuclear Power Plant (CCNPP) stores DSCs, each of which hold twenty-four old spent fuel assemblies. Each DSC contains an internal basket assembly, which includes twenty-four stainless steel guide sleeves (one for each spent fuel assembly), nine perforated carbon or stainless steel spacer discs, and four stainless steel support rods. The nine spacer discs are spaced out along the length of the DSC at locations that approximately coincide with the spent fuel assembly's eight spacer grids and single lower retention grid. The spacer discs are not structurally attached to the DSC shell walls or inner cover plates. The guide sleeves traverse the length of the DSC cavity through openings in the nine spacer discs. The support rods are used to maintain the spacer disc locations. The support rods traverse the length of the DSC cavity through the nine spacer discs, and are structurally welded to the spacer discs. The guide sleeves are attached only to the bottom support disc.

The DSC spacer discs, which are spaced so as to coincide with the fuel assembly's spacer grid plates, serve to provide rigid supports to the fuel assembly. When the DSC is in the horizontal position, the fuel assembly load is transferred through the spacer grid plates to the spacer discs. If the grid plates and the spacer discs are not properly aligned, then the grid plate that is offset from its respective spacer disc transfers the fuel load to the guide sleeve along its span between spacer discs. The guide sleeve, which is made of relatively thin stainless steel plate, will be subjected to the fuel assembly weight and horizontal drop loads. This leads to the concerns that the guide sleeve could get badly deformed or tear. In such an instance, the fuel rods would have a long improperly supported span and could get damaged. Also, the deformation of guide sleeve could hinder the fuel assembly retrievability.

Fuel assembly structural integrity evaluation was performed in Reference 5 with a conservative assumption, and is the subject of a separate safety evaluation. The horizontal drop evaluation was performed assuming that the guide sleeves were rigid. This is a conservative assumption when evaluating the spacer grids, which rest directly upon the guide sleeves, and other components of the fuel assembly. Any deformation of the guide sleeve due to grid misalignment will tend to soften the impact of the drop on the fuel assembly components. Therefore, the evaluation of structural integrity of the fuel assembly itself [Ref. 5] is not adversely impacted by the misalignment of spacer grids and spacer discs.

New Analysis

A new analysis (Section 4.7 of Reference 3) was performed to determine the impacts of misalignment of spacer discs with the spacer grid plates of the as-built fuel assemblies. The analysis focused on the guide sleeves, which are the vulnerable components, and the horizontal drop scenario, which would be the accident scenario with the most impact on the misaligned grids.

The guide sleeves are constructed of stainless metal plates bent into a sleeve with a square cross-section, and stitch-welded along one edge. The welds are 2 inches long at 6 inches center-to-center distance. The worst scenario was modeled in the analysis, which consisted of applying load to the non-welded span of the sleeve. The geometry analyzed covers the offsets at the LEF or the eighth zircalloy grid. The analysis was performed for the acceleration loads of 1g, 31g, and 75g.

It was concluded that the maximum deflection in the guide sleeve would be 0.271 inch, and the stresses generated in the sleeve would be larger, but still bounded by the stresses from the vertical drop scenario. The stresses would not be enough to cause any tearing of the sleeve material. Since the sleeve would not tear, fuel assemblies would remain retrievable.

The impact of the deformed sleeve would be on the inconel retention grid, inconel perimeter skirt/spacer grid, or the inconel perimeter skirt, depending upon the amount of offset in the alignment, but not on the fuel rods [Ref. 4]. Therefore, the fuel rods would not be damaged by the guide sleeve.

Reason for Activity:

This activity is being carried out in part to resolve Issue Reports IR3-007-609 and IR3-007-611 [Refs. 6 and 7]. The Issue Reports were written to address the misalignment of spacer discs and fuel assembly spacer grids. Spacer grid locations in fuel assemblies from batches 1/2D, 1/2E, 1/2F, 1/2G, 1/2H, 1J, 2A, 2B, and 2C were found not to coincide with the corresponding DSC spacer disc locations. The worst case misalignments consisted of the following:

- the mid-plane of the LEF flow plate (load bearing surface) being 0.9742 inch lower than the bottom surface of the bottom spacer disc, and
- the bottom of eighth zircalloy grid being 1.1017 inches lower than the DSC eighth spacer disc.

The design of the DSC is intended to ensure that the worst case postulated cask drop accident will not result in deformation of the DSC Internal Basket Assembly to such a degree that post-accident retrieval of intact fuel assemblies is not assured or that fuel rod integrity is compromised.

Technical Specification section 2.3, “Transfer Cask”, states that the transfer cask lifting height with a non-single-failure-proof lifting device shall not exceed 80 inches. The technical specification also states that for drops greater than 15 inches the DSC will be returned to the spent fuel pool and be visually inspected. Therefore, retrievability of fuel from the DSC needs to be assured.

Function(s) of affected SSC:

The affected systems, structures and components (SSCs) are DSCs and spent fuel assemblies. The DSCs consist of the outer canister and the internal basket assembly, which are classified as important-to-safety per 10 CFR 72. The subcomponents of the internal basket assemblies include the Spacer Discs, Support Rods, Guide Sleeves, and clip attachments. The spacer discs are spaced so as to coincide with the fuel assembly spacer grids.

The DSC provides containment, shielding, criticality control, configuration control related to fuel retrievability, structural support, and thermal safety functions during loading operations, transfer operations, and storage. It is designed to remain intact under all accident conditions identified in the ISFSI USAR with no loss of function. Specific design functions of the DSC include the following:

1. Confinement - The DSC design provides mechanical confinement of the stored fuel assemblies to prevent the dispersion of particulate or gaseous radionuclides from the fuel, and to maintain a barrier of helium around the fuel. The primary function of the DSC is to provide confinement of the spent nuclear fuel. This is achieved by the stainless steel shell and two inner cover plates (top and bottom ends) which are welded to the shell assembly. There are also outer cover plates (top and bottom) to further assure containment integrity. The DCS confinement boundary also is designed to retain helium cover gas inside the DSC in order to prevent corrosion of the fuel cladding and formation of expansive oxides in the fuel itself during storage. The confinement function is achieved by the outer canister portion of the DSC only, and hence is not affected by grid misalignment.
2. Criticality Control - The DSC design provides for sub-criticality during the wet loading, DSC drying, and interim storage operations. This is accomplished by a combination of mechanical separation of the fuel assemblies by the internal basket assembly and neutron absorption in the steel guide sleeve material.
3. Fuel Support and Configuration Control - The DSC internal basket assembly provides support for the spent fuel assemblies during normal operations. The DSC also provides configuration control related to post accident recovery of spent nuclear fuel. The DSC is designed so that the worst-case postulated accidents, including a cask drop, will not result in deformation of the Internal Basket Assembly or the DSC shell to such a degree that retrieval of intact fuel assemblies is not assured. The structural characteristics of the Transfer Cask (TC) and the DSC limit the deceleration loads on the fuel assemblies so that their integrity is assured in the worst-case drop accident.
4. Shielding - The DSC materials provide gamma radiation shielding. The DSC provides gamma shielding at its ends by the use of lead shield plugs. These provide ALARA dose rates at the top of the canister during drying and sealing operations and at the bottom for minimizing dose rates during DSC loading into the Horizontal Storage Module (HSM) and at the HSM door during storage. The shielding function is achieved by the outer canister portion of the DSC only, and hence is not affected by grid misalignment.
5. Thermal - Decay heat is removed by thermal radiation and conduction from the DSC to the TC, and by thermal radiation and conduction and convection from the DSC to the HSM. The DSC maintains the helium cover gas, which is required for corrosion control. This cover gas improves the thermal performance of the DSC. The decay heat removal function is achieved by the outer canister portion of the DSC only, and hence is not affected by grid misalignment.

Internal Basket Assembly - The functions of the internal basket assembly structures, systems and components are as follows:

1. Guide Sleeves – The guide sleeves establish storage compartments for 24 spent fuel assemblies within the DSC. The tops of the guide sleeves are flared to assist fuel handling operators in guiding the spent fuel assemblies into the sleeves. The guide sleeves are suspended approximately 1/2 inch by the clip attachments. This allows for DSC blowdown and drying via the DSC siphon port.
2. Spacer Discs – The spacer discs work together with the guide sleeves to maintain geometric separation of the fuel assemblies. The spacer discs support the weight of the guide sleeves, support rods and the spent nuclear fuel when the DSC is in a horizontal orientation. Because the spacer disc locations coincide with the fuel assembly spacer grids, for the most part, the guide sleeve by itself does not support the fuel assembly load.
3. Clip Attachments (Angles / Tabs) - The DSC internal basket assembly includes metal clip angle attachments or tab style clips, which are welded to the guide sleeves and the bottom spacer disc. The clips are designed as a fabrication convenience to restrain the guide sleeves during unloaded DSC transport, and during loading and unloading operations (fuel assembly insertion / extraction). The clip attachments also counteract withdrawal forces in the event of a stuck fuel assembly.
4. Support Rods – The support rods maintain the spacer disk location along the length of the DSC. They carry the weight of the guide sleeves, the clip angle attachments, and the spacer discs when the DSC is in a vertical orientation.
5. Fuel Assembly - The fuel assembly consists of 176 fuel and poison rods, 5 guide tubes, 5 guide tube sleeves, 8 fuel rod spacer grids, upper and lower end fittings, lower retention grid, and a hold-down device. The guide tubes, spacer grids, and end fittings form the structural frame of the assembly. The fuel rod spacer grids maintain the fuel rod pitch over the length of the assembly. The grid provides positive side restraint to the fuel rod but only frictional restraint axially. The spacer grids are the widest part on a fuel assembly. The four outer guide tubes are mechanically attached to the end fittings and the spacer grids are welded to all five guide tubes. The upper end fitting attaches to the guide tubes to serve as an aligning and lifting device for each fuel assembly. The spacer grids are the widest parts of the fuel assembly, and as such they come in contact with and transfer fuel assembly load to the guide sleeve.

SAR Revision No.: 8

SAR Sections Reviewed:

The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed are listed as follows:

1.3.1.1	Dry Shielded Canister Design
3.2.5.2	Dry Shielded Canister (Combined Load Criteria)
3.3.4.1	Control Methods for Prevention of Criticality
4.2.1.2	Dry Shielded Canister (Structural Specifications)
4.2.3.2	Dry Shielded Canister Description
5.1.1.9	Removal of Fuel from the Dry Shielded Canister
8.1.1.2	Dry Shielded Canister Analysis
8.1.1.3	Dry Shielded Canister Internal Basket Analysis
8.2.3.2	Accident Analysis
8.2.5	Cask Drop
8.2.12	Load Combinations
Table 3.6-3	Summary of Design Criteria for Accident Conditions
Table 8.1-3	Maximum Dry Shielded Canister Stresses for Normal Loads
Table 8.1-4	Maximum Dry Shielded Canister Stresses for Off-Normal Loads
Table 8.2-1	NUHOMS-24P Accident Loading Identification
Table 8.2-6	Maximum Dry Shielded Canister Stresses for Drop Accident Loads
Table 8.2-8	Dry Shielded Canister Enveloping Load Combination Results for Normal and Off-Normal Loads
Table 8.2-9	Dry Shielded Canister Enveloping Load Combination Results for Accident Loads, ASME Service Level C
Table 8.2-10	Dry Shielded Canister Enveloping Load Combination Results for Accident Loads, ASME Service Level D
Appendix A	Q:3.0-2, Load Combination and Design Criteria
Appendix A	Q:BGE001.0203 and Computer Run BPLRWZ, Spacer Disk Analysis

Tech Spec Bases Amendment/Rev No.: 2

Calvert Cliffs Independent Spent Fuel Storage Installation Technical Specifications, Appendix A to Materials License No. SNM-2505, Amendment 2, June 30, 2000

Tech Spec Bases Reviewed:

2.3	Transfer Cask (TC)
3/4.1	Fuel to be Stored at ISFSI
5.0	Design Features

Complete for 50.59 and 72.48:

1. **The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.**

☐ Yes ☒ No **May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Probability of Malfunction:

The proposed activity consists of evaluating misalignment of fuel assembly grids and DSC spacer discs. Spacer grid locations in fuel assemblies from batches 1/2D, 1/2E, 1/2F, 1/2G, 1/2H, 1J, 2A, 2B, and 2C were found not to coincide with the corresponding DSC spacer disc locations. The worst case misalignments consisted of the following:

- the mid-plane of the LEF flow plate (load bearing surface) being 0.9742 inch lower then the bottom surface of the bottom spacer disc, and
- the bottom of eighth zircalloy grid being 1.1017 inches lower than the DSC eighth spacer disc.

The SSCs affected by this activity consist of DSCs and spent fuel assemblies. In the DSCs, the affected subcomponents consist of the guide sleeves. Due to the misalignment, the fuel assembly load increases stresses on the guide sleeves. No other SSCs are affected.

During the normal handling and storage of DSCs, the forces exerted are equivalent to the accelerations of 1g to 2g. Section 4.7 of Reference 3 evaluated the misalignment issue for a cask drop scenario, with accelerations up to 75g. Results of the evaluation are discussed below under the Consequences of Accident. Based on the results it can be safely concluded that, for a loading of 1g to 2g, the probability of malfunction of the guide sleeves will not be increased, and therefore, the DSC and Internal Basket Assembly will be able to perform their functions.

☐ Yes ☒ No **May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Consequences of Malfunction:

The proposed activity will lead to a deformation of guide sleeves. However, as discussed above, the sleeve deformation is not large enough to prevent retrievability of fuel assemblies. Further, the sleeve deformation will not cause any damage to the fuel rods.

The proposed activity will not lead to breaching of the DSC barrier, or the loss of shielding, so that consequences of malfunction of equipment important to safety, namely the radiation dose to operators or radiation releases from ISFSI will not increase.

ACTIVITY: ES199800466	50.59 Log No.: N/A	72.48 Log No.: SE00148

ISFSI DSC Grid Misalignment

☐ Yes ☒ No **May the probability of occurrence of an accident previously evaluated in the SAR be increased?**

Probability of Accident:

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. They consist of loss of shielding, external missiles, earthquake, flood, cask drop, lightning, blockage of air inlets and outlets, DSC leakage, DSC overpressurization, and forest fire. Of these accidents, the cask drop accident and the earthquake incident are impacted by this activity, related to the misalignment of fuel assembly spacer grids and DSC spacer discs. The earthquake scenario is bounded by the cask drop accident, as the acceleration postulated in a design basis earthquake is 1.5g, which is much smaller than the acceleration in the drop accident of 75g. However, the probability of occurrence of cask drop, or any other accident is not increased by this activity.

This activity has no impact on the frequency, or on the probability of occurrence of design basis external natural events such as tornado, earthquake, or flood. Also, the proposed changes have no impact on the frequency, or on the probability of occurrence of design basis external man-induced events that could affect the ISFSI such as fires, or a Liquid Natural Gas (LNG) plant or pipeline spill or explosion.

Similarly, the misalignment of fuel assembly spacer grids and DSC spacer discs will have no impact on causes or initiating events for other accidents, such as blockage of air inlets and outlets, DSC leakage, or DSC overpressurization.

☐ Yes ☒ No **May the consequences of an accident previously evaluated in the SAR be increased?**

Consequences of Accident:

The proposed activity, namely validation of as-built misalignment of fuel assembly spacer grids and DSC spacer discs, will impact the cask drop accident and earthquake incident, as stated above.

The consequences of the cask drop accident with the misaligned grids were evaluated in Section 4.7 of Reference 3. The worst drop scenario was determined to be the horizontal drop. Deceleration rates of 31g and 75g were analyzed. The most vulnerable DSC component was the guide sleeve, which experiences higher stresses because of misalignment. The evaluation concluded that stresses generated in the sleeve would be larger, but still bounded by the stresses from the vertical drop scenario. The stresses would not be enough to cause any tearing of the sleeve material. The horizontal drop would cause a maximum deflection in the guide sleeve of 0.271 inch.

The impact on critical safety functions is discussed below. The critical functions affected will be configuration and criticality controls, and confinement. Other critical functions such as shielding and thermal safety are not affected by the misalignment of grids.

Criticality Control: The structural integrity of the fuel assembly was evaluated in Reference 5. The horizontal drop evaluation was performed assuming that the guide sleeves are perfectly rigid, which is conservative for determining the impact on the structural integrity of the fuel assembly itself. Therefore, the fuel evaluation is not affected by the misalignment of spacer grids and spacer discs, and hence, the criticality control function is not impacted by the misalignment of grids.

Configuration Control: Configuration of fuel assemblies within the DSC needs to be maintained such that the assemblies remain retrievable. Since it was determined that the guide sleeve would not tear, fuel assemblies would remain retrievable.

Confinement: The worst case misalignment would cause a maximum deflection in the guide sleeve of 0.271 inch. The impact of the deformed sleeve would be on the inconel retention grid, inconel perimeter skirt/spacer grid, or the inconel perimeter skirt, depending upon the amount of offset in the alignment, but not on the fuel rods [Ref. 4]. Therefore, the fuel rods would not be damaged by the guide sleeve, and the radioactive fission products would remain confined.

Shielding: The DSC Internal Basket Assembly is not credited with augmenting the radiation shielding properties of the DSC, and there is no direct interface between the Internal Basket Assembly Components, and the DSC shielding materials. Therefore, there is no impact on the DSC shielding due to this activity.

Thermal Control: The parameters affecting the thermal analysis, such as heat generation or transfer capabilities are not impacted by the proposed activity. Therefore, there is no impact on the DSC thermal control due to this activity.

2. **The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.**

☐ Yes ☒ No **May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?**

Possibility of New Malfunction:

The proposed activity validates the misalignment between fuel assembly spacer grids and DSC spacer discs. The most impacted component is the guide sleeve. The evaluation showed that the sleeve would experience higher stresses, but within the allowable values [Ref. 3]. Therefore, possibility of a malfunction of a different type is not created.

The other impacted SSCs, namely the spacer discs and support rods continue to function as per their design. They continue to provide the structural support to maintain the fuel assembly configuration within the DSC, and keep the fuel assemblies retrievable. The possibility of a malfunction of a different type than any previously evaluated is not created.

☐ Yes ☒ No **May the possibility of an accident of a different type than any previously evaluated in the SAR be created?**

Possibility of New Accident:

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the USAR, and have been discussed previously. Reanalysis with the misaligned grids showed an increase in the stress level in the guide sleeve during a top drop accident [Ref. 3]. However, since there is no change to the design or operation of the NUHOMS system caused by this activity, the possibility of an accident of a different type than any previously evaluated in the SAR will not be created.

3. **The margin of safety as defined in the basis for any Technical Specification is not reduced.**

☐ **Yes** ☒ **No** **Will the margin of safety as defined in the basis for any Technical Specification be reduced?**

Tech Spec Bases 2.3

Discussion of why the margin of safety is not reduced:

Technical Specification Section 2.3, "Transfer Cask", states that the transfer cask lifting height with a non-single-failure-proof lifting device shall not exceed 80 inches, since analyses performed for the DSC and transfer cask confirm that drops of 80 inches could be sustained without unacceptable damage or without decreasing margins of safety. In addition, the technical specification also states that for drops greater than 15 inches the DSC will be returned to the spent fuel pool and be visually inspected. Therefore, retrievability of fuel from the DSC needs to be demonstrated.

Analysis demonstrated that the misalignment of grids would cause the DSC guide sleeve to deform during a horizontal drop accident, reducing the gap between the guide sleeve and the fuel assembly. However, the amount of deflection would be less than the gap, therefore, the fuel assembly would not be pinched. This deformation will not affect fuel assembly retrievability. Also, the fuel rods would not be damaged from the guide sleeve deformation.

Another scenario that could affect fuel assembly retrievability would be piercing of the guide sleeve at the lower end fitting. The analysis showed, however, that the guide sleeves would not be pierced.

The margin of safety is the difference between the allowable stress value based on ASME B&PV Code, Section III, and the material equivalent failure stress. Reference 3 concluded that the computed stress levels are below the allowable stresses. The analysis demonstrated that there would be no unacceptable damage to the DSC.

Therefore, since the fuel assembly can still be retrieved after a horizontal drop accident, computed stresses remain below the allowables, and there is no impact to the integrity of the fuel rods, the margin of safety is not reduced.

Complete for 72.48:

☐ **Yes** ☒ **No** **Will the proposed activity involve a significant increase in occupational dose?**

A significant increase in occupational dose:

The design and operation of the NUHOMS-24P dry cask storage system is not changed by this activity. This activity does not reduce the integrity of the confinement boundary, and the radioactive shielding elements are not affected. Retrievability of fuel following a design basis cask drop accident is not impacted by this activity. The fuel rods are not damaged. Because none of these attributes are changed, the occupational doses summarized in USAR Table 7.4-1 are not affected by this activity. Therefore, no occupational doses are increased.

SAFETY EVALUATION FORM

ACTIVITY: ES199800466	50.59 Log No.: N/A	72.48 Log No.: SE00148

ISFSI DSC Grid Misalignment

☐ Yes ☒ No **Will the proposed activity involve a significant unreviewed environmental impact?**

A significant unreviewed environmental impact:

The NUHOMS-24P dry cask storage system confinement and radiological shielding functions are not reduced by this activity. The DSCs with misaligned fuel assembly spacer grids and DSC spacer discs, have been evaluated [Ref. 3] to ensure stresses will remain within allowable limits under the most severe postulated drop accident conditions.

This activity does not affect any area of the plant site previously undisturbed for the ISFSI, and does not cause any reason for revision to the ISFSI Updated Environmental Report. This activity does not affect the environmental conditions associated with the ISFSI. Therefore, this activity does not involve an unreviewed environmental impact.

References:

1. Calvert Cliffs Independent Spent Fuel Storage Installation USAR, Rev. 8
2. UFSAR/USAR/TSB Change Request, UCR No. 98-020, for ESP ES199800466-000
3. BGE Calculation CA04132, Rev. 0002
4. Hopper and Associates Letter HABGE-04/99-0778, from Kenneth L. Saunders to James E. Remeniuk, 4/14/99
5. BGE Calculation CA04680, Rev. 0000, C-E 14X14 Fuel Grid Horizontal Drop Evaluation
6. BGE Issue Report IR3-007-609, 03/24/1997
7. BGE Issue Report IR3-007-611, 03/24/1997
8. Technical Specifications for Calvert Cliffs ISFSI, Amendment 2
9. Calvert Cliffs ISFSI Updated Environmental Report, Rev. 1
10. 10CFR72.48 Safety Evaluation No. SE00133, ISFSI – Analysis of DSC Under Postulated Cask Drop Accident

SAFETY EVALUATION FORM

ACTIVITY: ES199800466	**50.59 Log No.: N/A**	**72.48 Log No.: SE00148**

ISFSI DSC Grid Misalignment

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: The proposed activity consists of making changes to the ISFSI USAR, in order to account for the misalignment of some of the fuel assembly spacer grids in relation to the spacer discs provided within the Dry Shielded Canister (DSC). The USAR Section 1.3.1.1 states that the DSC spacer disks are "at intervals corresponding to the fuel assembly spacer grids". This was found not to be the case for a few of the spacer grids of fuel assemblies from batches 1/2 D/E/F/G/H, 1J, 2A, 2B and 2C. Misalignment was found in fuel assembly lower end fittings (LEFs) and the DSC bottom spacer discs; and in fuel assembly eighth zircalloy grids and the DSC eighth spacer discs.

The changes being made to the ISFSI UFSAR consist of revisions to Sections 1.3.1.1 and 8.2.5.2 to add statements regarding the misalignment of grids of fuel assemblies and DSCs, and its analysis. Stress values reported in Tables 8.2-6 and 8.2-10 envelope the stresses from misalignment configuration, therefore no changes are necessary.

Reason for Activity: This activity is being carried out in part to address the misalignment of spacer discs and fuel assembly spacer grids, as described above.

The design of the DSC is intended to ensure that the worst case postulated cask drop accident will not result in deformation of the DSC Internal Basket Assembly to such a degree that post-accident retrieval of intact fuel assemblies is not assured or that fuel rod integrity is compromised.

Technical Specification Section 2.3, “Transfer Cask”, states that the transfer cask lifting height with a non-single-failure-proof lifting device shall not exceed 80 inches. The technical specification also states that for drops greater than 15 inches the DSC will be returned to the spent fuel pool and be visually inspected. Therefore, retrievability of fuel from the DSC needs to be assured.

Activity Summary: A new analysis was performed to determine the impacts of misalignment of spacer discs with the spacer grid plates of the as-built fuel assemblies. The analysis focussed on the guide sleeves, which are the vulnerable components, and the horizontal drop scenario, which would be most impact by the misaligned grids.

The guide sleeves are constructed of stainless metal plates bent into a sleeve with a square cross-section, and stitch-welded along one edge. The welds are 2 inches long at 6 inches center-to-center distance. The worst scenario was modeled in the analysis, which consisted of applying load to the non-welded span of the sleeve. The geometry analyzed covers the offsets at the LEF or the eighth zircalloy grid. The analysis was performed for the acceleration loads of 1g, 31g, and 75g.

It was concluded that the stresses generated in the guide sleeve would be larger, but still bounded by the stresses from the vertical drop scenario. The stresses would not be enough to cause any tearing of the sleeve material. Since the sleeve would not tear, fuel assemblies would remain retrievable.

The analysis determined that a small deflection in the guide sleeve would occur. The deformed sleeve would have an impact on the inconel retention grid, inconel perimeter skirt/spacer grid, or the inconel perimeter skirt, depending upon the amount of offset in the alignment, but not on the fuel rods. Therefore, the fuel rods would not be damaged by the guide sleeve.

USQ Determination: This activity was evaluated against the criteria of 10CFR72.48(a)(2), such as the probability of occurrence or the consequences of an accident or the malfunction of equipment important to safety, and it was concluded that it does not involve an unreviewed safety question (USQ).

ATTACHMENT 3, SAFETY EVALUATION FORM

Page 1 of 7

ACTIVITY: ES199801283-001 Rev 0	**50.59 Log No.:** N/A	**72.48 Log No.: SE00150**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ **YES** ☒ **NO** **Involve an unreviewed safety question (USQ)?**
☐ **YES** ☒ **NO** **Involve a change in the Technical Specifications/License Conditions?**
☒ **YES** ☐ **NO** **Require a change or addition to the UFSAR/[✓ USAR]/Technical Specification Bases?**

Applicable to 10 CFR 72.48 Safety Evaluations

☐ **YES** ☒ **NO** **Involve a Significant Increase in Occupational Dose?**
☐ **YES** ☒ **NO** **Involve a Significant Unreviewed Environmental Impact?**

Prepared by: Carl Faller **Department:** CEU **Date:** 10/14/99
PRINTED NAME AND SIGNATURE
J.E. REMENIUK 10-14-99

☐ **YES** ☒ **NO** **Is a special review required by groups other than the group to which the Preparer belongs?**

Resp Ind.: D M DEAN PRINTED NAME	**Resp Ind.:** R.M. STARK PRINTED NAME	**Resp Ind.:** N/A PRINTED NAME
SIGNATURE	SIGNATURE	N/A SIGNATURE
Work Group: SECURITY	**Work Group:** EEU	**Work Group:** N/A
Date: 10/25/99	**Date:** 10/25/99	**Date** N/A

Approved ☑ **Disapproved** ☐	**Approved** ☑ **Disapproved** ☐
Signature G.V. PATEL INDEPENDENT REVIEWER	**Signature** PRAVIN PATEL Pravin Patel GS-DES, GS-TES, or PE-PDSU
Date 10-15-1999	**Date** 10-25-1999

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 99-085 **Date:** 11-1-99

Recommend Approval ☑	**Recommend Disapproval** ☐	**Signature** **Date** 11-1-9 POSRC CHAIRMAN
Approved ☑	**Disapproved** ☐	**Signature** **Date** 11/1/99 PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ **YES** ☒ **NO**
Signature:
OSSRC SES Chairman
Date: 1/20/00

If yes, OSSRC Meeting No.

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199801283-001 Rev 0	**50.59 Log No.: N/A**	**72.48 Log No.: SE00150**

Proposed Activity:
1) Installation of a permanent personnel gate (locked) in the west side of the nuisance perimeter fence (the outer fence) surrounding the Independent Spent Fuel Storage Installation (ISFSI)

2) Installation of a temporary security fence (with a locked vehicle gate), temporary nuisance barrier, and temporary remote sensing system (infrared). These will be oriented in the east / west direction located approximately 27 ft, 47ft, and 37 ft respectively south of existing Horizontal Storage Modules (HSM) 2A and 2B.

The above activities deviate from ISFSI USAR Figure 1.2-1 and 4.1-2 and therefore require a safety evaluation. Concurrent with the physical fence changes, the following administrative changes will be made as well:

1) Correction of HSM number scheme on ISFSI USAR Figure 2.4-1 to reflect "As Built"
2) Correction of vehicle gate location on ISFSI USAR Figures 1.2-1 and 4.1-2 to more accurately reflect the "As Built" location.
3) Changes to reflect the addition of HSM 3A and 3B to the ISFSI, as authorized and screened under engineering package ES199801283-000.

Reason for Activity:
HSM 3A and 3B are being added to the ISFSI. Construction of 3A and 3B will require the daily admittance of construction personnel inside the fenced ISFSI protected area. Doing so allows such individuals access to installed and fuel loaded HSM1A, 1B, 2A, and 2B which in turn requires increased security personnel to manage these individual's activities.

By installing a temporary fence, nuisance barrier, and remote security detection system south of the existing HSM 2A and 2B, the current ISFSI protected area boundary can be collapsed to an area more immediately surrounding HSM 1A, 1B, 2A, and 2B. This will ease the number of security personnel required to monitor construction personnel activities and will allow more freedom of movement for the construction personnel themselves.

The permanent personnel gate in the west side of the nuisance perimeter fence surrounding the ISFSI is being added at Nuclear Security's request in order to allow them access flexibility. Currently access to the area between the perimeter nuisance fence (the outer fence) and the security fence (the inner fence) where the remote sensing devices are located is through one location only; the vehicle-sized gate located on the east side. Access to the overall ISFSI protected area will still be via one location only, the existing east side vehicle size access gate.

Function(s) of affected SSC:
The primary function of the double perimeter fence arrangement (with remote sensing devices) that surrounds the ISFSI is for security purposes; to restrict unauthorized personnel immediate and easy access to the HSMs. The fence system components function as follows:

a) The function of the outer 8-foot high chainlink fence is to keep people and animals (e.g. deer) that near the ISFSI from inadvertently triggering the remote sensing devices. This fence is commonly referred to as the "Nuisance Fence"
b) The function of the inner 12-foot high chainlink fence is to keep unauthorized people from accessing the HSMs. This fence is the main security fence.
c) The function of the remote sensing devices located between the security and nuisance fences is to alert security when unauthorized personnel breech the nuisance fence and approach the inner security fence.

A secondary function of the fence system is to reduce the potential for blockage of HSM air inlets and outlets from debris thrown about by environmental events such as tornadoes. The fence was not designed to withstand any particular size or amount of debris traveling at any particular speed but credit is taken for its presence and ability in reduce the potential that debris outside the fence may get inside the ISFSI fenced area and reach HSM inlet and outlet air vents.

ACTIVITY: ES199801283-001 Rev 0	**50.59 Log No.:** N/A	**72.48 Log No.: SE00150**

SAR Revision No.: ISFSI USAR Rev 8 . **SAR Sections Reviewed:** ISFSI Sections 2.1.2 "Site Description" Chapter 3 "Principal Design Criteria" 3.3.5 "Radiological Protection" 8.2.7 "Blockage of Air Inlets and Outlets"	**Tech Spec Bases Rev. No:** ISFSI Tech Spec Rev 0 Amendment 1 **Tech Spec Bases Reviewed:** 2.0 to 2.4 and 3 / 4.1 to 3 / 4.5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ YES ☒ NO **May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Probability of Malfunction:

The accidents discussed in the ISFSI USAR Chapter 8 are:
Loss of Air Outlet Shielding, Tornado Winds / Missiles, Earthquake, Flood, Cask Drop, Lightning, Blockage of Air Inlets and Outlets, Dry Shielded Canister Leakage, Accidental Pressurization of Dry Shielded Canister, Forest Fire, Liquefied Natural Gas Plant or Pipeline Spill or Explosion, Load Combinations, Other Event Considerations (Storage of Flammable Liquid Fuel)

The fence perimeter system is a passive system that restricts personnel access to the HSMs. It is not a major contributor to prevent or mitigate any of the above accidents. It is discussed in section 8.2.7.1 of the USAR and is mentioned as a contributor in reducing the potential of tornado debris from blocking the HSM air inlet and outlets.

The permanent personnel gate will still provide the above functions. The main perimeter fence is not being removed with the installation of the temporary fence so it is also available to provide these functions. Additionally, the temporary fence functions to reduce the potential that construction debris inside the ISFSI and south of the temporary fence will enter into the existing HSM area.

The changes in HSM numbering and reflection of actual location of the east vehicle size security gate are administrative in nature.

The proposed changes to the ISFSI USAR do not constitute an increase in probability of a malfunction.

☐ YES ☒ NO **May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Consequences of Malfunction:

The fence perimeter system does not significantly mitigate any accidents discussed in the ISFSI USAR Chapter 8 therefore changes to the fence system do not increase any malfunction consequences. Failure of the permanent or temporary fence during a tornado introduces the same and equivalent circumstances as the original fence system.

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199801283-001 Rev 0	**50.59 Log No.: N/A**	**72.48 Log No.: SE00150**

☐ YES ☒ NO **May the probability of occurrence of an accident previously evaluated in the SAR be increased?**

Probability of Accident:

The fence is a passive system and does not actively interact with the HSMs. Because there is no interaction, permanent and temporary changes to the fence do not alter any ISFSI USAR Chapter 8 accident frequencies or increase the probability of any accidents.

The changes do not alter the fence's function to reduce potential tornado affects discussed in "Blockage of Air Inlets and Outlets" because the changes do not make the probability of a tornado occurrence higher.

The changes in HSM numbering and reflection of actual location of the east vehicle size security gate are administrative in nature.

☐ YES ☒ NO **May the consequences of an accident previously evaluated in the SAR be increased?**

Consequences of Accident:

The proposed fence changes do not change, degrade or prevent actions described in or assumed in the ISFSI USAR Chapter 8 accidents.

The ability of the fence to reduce potential tornado affects (i.e. debris blocking HSM air inlets and outlets) remains intact after the changes are made. The original perimeter fence will remain in place and the permanent personnel gate being installed serves to block tornado debris in an equivalent method to that of the main fence.

The changes to the fence do not alter any assumptions previously made in evaluation of radiological consequences of Chapter 8 accidents because the changes do not alter the location of the ISFSI with relation to BGE controlled area boundaries. Therefore off site doses from accidents are not affected.

The fence is passive and does not play a direct role in mitigating radiological consequences of Chapter 8 accidents.

The fence alterations do not affect any fission product barriers.

The changes in HSM numbering and reflection of actual location of the east vehicle size security gate are administrative in nature.

ACTIVITY: ES199801283-001 Rev 0	**50.59 Log No.: N/A**	**72.48 Log No.: SE00150**

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ **YES** ☒ **NO** **May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?**

Possibility of New Malfunction:

The changes to the fence do not pose the possibility of a new malfunction because it is a passive component of the ISFSI and does not physically interact with the HSMs. The fence does not have any active attributes that could prevent an ISFSI system from mitigating the consequences of a Chapter 8 accident nor are there any indirect means the fence could create a new malfunction.

The changes in HSM numbering and reflection of actual location of the east vehicle size security gate are administrative in nature.

☐ **YES** ☒ **NO** **May the possibility of an accident of a different type than any previously evaluated in the SAR be created?**

Possibility of New Accident:

The changes to the perimeter fence do not create any new credible accidents not already discussed in the ISFSI USAR Chapter 8. The main reason there is no impact is because the fence is a passive ISFSI component and does not interact directly with the HSMs. The fence changes cannot contribute to changes in fuel storage temperatures, changes to the cooling means of the stored canisters, increase the affects of natural events, impact the movement of canisters, affect the sealing methods of the storage canisters, affect off sight doses, etc. .

The changes in HSM numbering and reflection of actual location of the east vehicle size security gate are administrative in nature.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ **YES** ☒ **NO** **Will the margin of safety as defined in the basis for any Technical Specification be reduced?**

Bases	**Discussion of why the margin of safety is not reduced**
2.1 to 2.4 “ Functional and Operating Limits” 3 / 4.1 to 3 / 4.5 “LCO /SRs”	None of the temperature limits, irradiation limits, or radiation limits. discussed in the bases depend directly or indirectly on the location or function of the perimeter fence or administrative changes being made to the ISFSI figures

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199801283-001 Rev 0	**50.59 Log No.: N/A**	**72.48 Log No.: SE00150**

Complete for 72.48:

☐ **YES** ☒ **NO** **Will the proposed activity involve a significant increase in occupational does?**

A significant increase in occupational dose:

None of the "operation" activities listed in the ISFSI USAR Table 7.4-1 "Estimated Occupational Exposure for One Horizontal Storage Module Load" are impacted by the installation of the permanent gate or the temporary fence.

The permanent west gate is a personal access gate and does not interact with cask loading evolutions because it only permits access to the area between the nuisance fence (the outer fence) and the security fence (the inner fence). The dose estimates for Table 7.4-1 activities are therefore not affected by this change.

The temporary fence will be located 27 feet south of HSM 2A and 2B which does not leave enough room to install casks on the south side as long as the temporary fence is in that particular location. If a future decision is made to shift the fence more southerly to accommodate loading the south side 2A and 2B HSMs while the temporary fence is installed, the Table 7.4-1 activities will still not be significantly affected. The "operation" activities described in this Table 7.4-1 all take place in close proximity to the cask and the HSM itself. The fence will not slow these "operation" to any degree that an increase in dose to personnel will result.

The changes in HSM numbering and reflection of actual location of the west security gate are administrative in nature.

☐ **YES** ☒ **NO** **Will the proposed activity involve a significant unreviewed environmental impact?**

A significant unreviewed environmental impact:

The fence changes do not alter any plant area footprints already dedicated for ISFSI installation or support.

A revision to the ISFSI Environmental Impact Statement is not required by these activities.

Summary: (For NRC Report, provide a brief overview)

This Safety Screen evaluates the following changes to the USAR of the Independent Spent Fuel Storage Installation (ISFSI) located at Calvert Cliffs Nuclear Power Plant:

1) The acceptability to install a new permanent personnel gate (locked) in the west side of the nuisance perimeter fence (the outer fence) surrounding the ISFSI.

2) The acceptability to install a temporary security fence (with a locked vehicle gate), nuisance barrier, and remote infrared security system. This temporary fence system will be oriented in the east / west direction located approximately 27 ft, 47ft, and 37 ft respectively south of existing Horizontal Storage Module (HSM) 2A and 2B.

3) Correction to the number scheme of the HSMs (administrative in nature)

4) Correction to the physical location of the existing vehicle gate in the east side of the perimeter fence (administrative in nature)

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199801283-001 Rev 0	**50.59 Log No.: N/A**	**72.48 Log No.: SE00150**

Reason:
Calvert Cliffs is adding HSMs 3A and 3B to the ISFSI (Reference Engineering Package ES199801283-000). Construction of 3A and 3B will require the daily admittance of construction personnel inside the fenced ISFSI protected area. Doing so allows such individuals access to existing and fuel loaded HSM 1A, 1B, 2A, and 2B which in turn requires increased security personnel to manage these individuals' activities. To manage this situation, a temporary fence, nuisance barrier, and remote sensing system will be installed just south of HSM 2A and 2B so the ISFSI protected area boundary can routinely be collapsed during construction to an area more immediately surrounding HSM 1A, 1B, 2A, and 2B. This will ease the number of security personnel required to monitor construction personnel activities and will allow more freedom of movement for the construction personnel themselves. The permanent personnel gate in the west side of the nuisance perimeter fence surrounding the ISFSI is being added at Nuclear Security's request in order to allow them access flexibility. Currently access to the area between the perimeter nuisance fence (the outer fence) and the security fence (the inner fence) where the remote sensing devices are located is through one location only; the vehicle-sized gate located on the east side. Access to the overall ISFSI protected area will still be via one location only, the existing east side vehicle size access gate.

The above activities deviate from information currently reflected in the ISFSI USAR Figures 1.2-1, 4.1-2, and 2.4-1 and therefore require a Safety Evaluation.

These changes do not represent an Unreviewed Safety Question (USQ), a Significant Increase in Occupational Dose, or an Unreviewed Environmental Impact.

The perimeter fencing system is a passive system surrounding the HSMs whose main function is security (i.e. to prevent unauthorized personnel from accessing the HSMs). Except for "Blockage of Air Inlets and Outlets" the fence is not credited in any ISFSI USAR Chapter 8 accident analysis, prevention assumptions, or mitigations. In the case of "Blockage of Air Inlets and Outlets" the perimeter fence along with the HSM air inlet and outlet physical separation as credited as a contributor to "reducing the potential" that the vents will become blocked by any debris stirred up by a tornado. No calculations, assumptions, or credit was taken for the fence stopping any particular size or amount of debris traveling at any particular speed. It is just referenced as being present.

No USQ results from:
- Installation of the permanent personnel gate because it will be constructed of equivalent material, size, and location as the existing fence.
- The temporary fence system because the original perimeter fence will remain in place to reduce the potential of debris from entering the ISFSI. Any debris bounded inside the ISFSI but south of the temporary fence (i.e. the construction area of HSM 3A and 3B) will be reduced from reaching the existing HSMs by the equivalent temporary fence.
- Changes to the HSM number system or figure change to shown the actual location of the existing security gate (east side) as these are administrative in nature and do not cause any major deviations from the ISFSI USAR.

A Significant Increase in Occupational Dose to ISFSI USAR Table 7.4-1 does not occur because the fence changes do not affect the activities listed in Table 7.4-1

A Significant Unreviewed Environmental Impact does not result from these changes because the footprint to the ISFSI is not being altered and the Environmental Impact Statement requires no changes.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 1 of 4)

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ACTIVITY: **Organization Change** **50.59 Log No.:** SE00409 **72.48 Log No.:** SE 00151

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Michael A. Cox Michael A. Cox Department: 47 Date: 12/1/99
PRINTED NAME AND SIGNATURE

☐ YES ☒ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: ______ **PRINTED NAME**	Resp. Ind.: ______ **PRINTED NAME**	Resp. Ind.: ______ **PRINTED NAME**
______ **SIGNATURE**	______ **SIGNATURE**	______ **SIGNATURE**
Work Group: ______	Work Group: ______	Work Group: ______
Date: ______	Date: ______	Date: ______

Approved ☑ Disapproved ☐
Signature LMDaniels LMDaniel
INDEPENDENT REVIEWER
Date 12/15/99

Approved ☒ Disapproved ☐
Signature
GS-DES, GS-TSES, or PE-PDSU
Date 12/15/99

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 99-101 Date: 12-22-99

Recommend Approval ☑ Recommend Disapproval ☐ Signature ______ Date 12-22-99
POSRC CHAIRMAN

Approved ☑ Disapproved ☐ Signature ______ Date 12/28/99
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ YES ☑ NO
Signature: W Russell Date: 5/18/00
OSSRC SES Chairman
If yes, OSSRC Meeting No. ______

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 2 of 4)

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ACTIVITY: **Organization Change** **50.59 Log No.:** **SE00409** **72.48 Log No.:** **SE00151**

Proposed Activity: The proposed activity is an organization change that establishes the Integrated Work Management Section. The new Master Section reports to the Plant General Manager. The following three sections will report to the Integrated Work Management Master Section: 1)Integrated Planning/Risk Management Section, 2) Outage Management Section, and 3) Scheduling/Work Coordination Section. This ~~or~~ change will require a revision to Chapter 12 in the UFSAR.

Reason for Activity: To support the establishment of an Integrated Work Management Program

Function(s) of affected SSC: none

SAR Revision No.: 26

Tech Spec Bases Amendment/Rev No.: N/A

SAR Sections Reviewed: Chapter 12

Tech Spec Bases Reviewed: N/A

Complete for 50.59 and 72.48:

I. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ Yes ☒ No May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: The probability of a malfunction of equipment will not be increased. The new organization will work with line organizations to use disciplined methodologies to manage the risk associated with work activities.

☐ Yes ☒ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: The consequences of a malfunction will not be increased. Since the new organization is aimed at properly managing risk associated with work activities, work activities will be planned to avoid or mitigate the risks associated with performing work on equipment. These actions will NOT increase consequences of malfunctions.

☐ Yes ☒ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: The proposed organizational change helps improve Calvert Cliffs ability to manage the risk associated with work activities. As a result of improved risk management, the probability of an accident will NOT increase.

☐ Yes ☒ No May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident: Since the new organization is aimed at properly managing risk associated with work activities, work activities will be planned to avoid or mitigate the risks associated with performing work on equipment. Improved risk management is independent of Consequences of a Chap 14 accident.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 3 of 4)

Page 3 of 4

ACTIVITY **Organization Change** **50.59 Log No.:** **SE00409** **72.48 Log No.:** **SE00151**

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ Yes ☒ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction: By assessing risk and improving the way we manage that risk, we will only have a decrease in the possibility of any malfunctions.

☐ Yes ☒ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident: Improved risk mangement and integrated work management is aimed at reducing risk associated with work activities. This includes nuclear safety risk which is a significant contributor to the possibility of an accident. Since nuclear safety risk management will NOT decrease, the possibility of an accident will NOT increase.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ Yes ☒ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
	The proposed organziation change has no direct impact on the margin of safety as defined in any Tech Specs.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 4 of 4)

Page 4 of 4

ACTIVITY: **Organization Change** **50.59 Log No.:** **SE00409** **72.48 Log No.:** **SE00151**

Complete for 72.48:

☐ Yes ☒ No Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose: Integrated work management and improved risk management will help to avoid or minimize occupational exposure associated with ISFSI or other work activities which have radiation exposure risk.

☐ Yes ☒ No Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact: The new organization's improved risk management procedures recognize environmental risk and seeks to mitigate that risk so that no Unreviewed Environmental Questions are introduced.

Summary: (For NRC Report, provide a brief overview)

The proposed activity establishes an Integrated Work Management organization under the Plant General Manager. The new organization will improve risk management activities by institutionalizing the way we assess and manage risk in the following areas: 1) Nuclear Safety 2) Industrial Safety 3) Radiation Safety 4) Environmental Safety and 5) Corporate Risk.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 1 of 8)

ACTIVITY: **ES199701539-002** 50.59 Log No.: 72.48 Log No.: **SE00152**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: IM Sommerville Department: Nuclear Date:
PRINTED NAME AND SIGNATURE

☒ YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: R.H. Beall **PRINTED NAME**	Resp. Ind.: J.E. Remeniuk **PRINTED NAME**	Resp. Ind.: **PRINTED NAME**
SIGNATURE	**SIGNATURE**	**SIGNATURE**
Work Group: **NFM**	Work Group: **CEU**	Work Group:
Date: 4/6/00	Date: 4.27.00	Date:

Approved ☑ Disapproved ☐	Approved ☒ Disapproved ☐
Signature Mark Finley **INDEPENDENT REVIEWER**	Signature **GS-DES, GS-TSES, or PE-PDSU**
Date 5/4/2000	Date 5/15/00

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 00-047 Date: 5/15/00

Recommend Approval ☑ Recommend Disapproval ☐ Signature Date 5/15/00
POSRC CHAIRMAN

Approved ☑ Disapproved ☐ Signature Date 5/16/00
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ YES ☑ NO

Signature: Date: 7/14/2000
OSSRC SES Chairman

If yes, OSSRC Meeting No. ____________

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 2 of 8)

ACTIVITY: **ES199701539-002** 50.59 Log No.: 72.48 Log No.: **SE00152**

Proposed Activity: See Attached Sheets

Reason for Activity: See Attached Sheets

Function(s) of affected SSC: See Attached Sheets

SAR Revision No.: 8	Tech Spec Bases Amendment/Rev No.: 1
SAR Sections Reviewed: 3.2, 8.2	Tech Spec Bases Reviewed: B2.1

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ Yes ☒ No May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: See Attached Sheets

☐ Yes ☒ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: See Attached Sheets

☐ Yes ☒ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: See Attached Sheets

☐ Yes ☒ No May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident: See Attached Sheets

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ Yes ☒ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction: See Attached Sheets

☐ Yes ☒ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident: See Attached Sheets

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 3 of 8)

ACTIVITY: **ES199701539-002** 50.59 Log No.: 72.48 Log No.: **SE00152**

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ Yes ☒ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

See Attached Sheets

Complete for 72.48:

☐ Yes ☒ No Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

☐ Yes ☒ No Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

Summary: (For NRC Report, provide a brief overview)

See Attached Sheets

72.48 - SAFETY EVALUATION
ISFSI - FOREST FIRE RE-ANALYSIS

PROPOSED ACTIVITY

Change the ISFSI SAR, Section 8.2.10.2, to reflect the modified analysis of the Forest Fire Accident.

Reason for Activity

During review of the Forest Fire accident analysis, an error was discovered. This error had an effect on the temperatures experienced by the concrete of the Horizontal Storage Module (HSM) during the fire. An Issue Report, IR1-043-510, was written to track resolution and the Forest Fire analysis was revised, CA03945, Rev 1. This 72.48 discusses the results of the analysis.

Function of the Affected SSC

The Dry Shielded Canister (DSC) has containment, shielding, criticality control, and thermal safety functions. The primary function of the DSC is to provide containment for the spent nuclear fuel. This is achieved by the stainless steel shell and two inner cover plates (top and bottom ends) which are welded to the shell assembly. There are redundant outer cover plates (top and bottom) to assure containment integrity. The DSC provides gamma shielding at its ends by the use of thick steel end plugs. These provide ALARA dose rates at the top of the canister (for DSC drying and sealing operations) and at the bottom (for minimizing dose rates at the HSM doorway). Shielding in the radial direction is not a safety function of the DSC, although it does provide a small amount due to the shell thickness.

Criticality control is provided by the DSC's internal basket assembly. A series of spacer disks and axial support rods maintain the fuel assemblies in known positions under all normal and accident conditions. The thickness and location of the spacer disks plus the relative locations of the fuel assemblies achieve the criticality control function. The DSC maintains the helium cover gas which is required for heat rejection and corrosion control. Heat is transferred via thermal radiation and conduction from the fuel through the guide sleeves, spacer disks, and cover gas to the DSC shell, where it is convectively cooled during HSM storage. The fuel rod cladding serves as a primary confinement boundary for the fuel pellet and fission products. The preservation of the fuel cladding integrity is intended to prevent oxidation of the Uranium fuel material.

During storage, the horizontal storage module supports the DSC, provides it with protection from weather and external missiles, permits heat rejection in order to maintain concrete temperatures and fuel cladding temperatures to within design allowables, and protects the public and operating personnel from radiation which emanates from the DSC. During fuel loading, the horizontal storage module additionally provides a stable surface for docking the transfer cask or a compatible shipping cask and performing DSC transfers. The HSM is designed to withstand DSC handling loads (normal and off-normal), environmental extremes (heat, rain, snow, wind), earthquakes, lightning, external missiles (tornado, man-made), floods, fires, and explosions.

Revised Forest Fire Analysis

The accident analysis for the Forest Fire has been modified with BGE calculation CA03945, Rev 0001. This analysis revision corrects errors in the original analysis associated with radiation heat transfer view factors and initial HSM wall temperatures. The resulting analysis indicates that the temperatures expected are higher than those previously calculated. The calculation revision establishes criteria for evaluation of the results and evaluates the results against these criterion to confirm adequacy of the resulting fire effects on the HSM and DSC.

Criteria specified includes: allowable concrete temperature, fuel cladding temperature, HSM structural analysis, DSC internal pressure and post fire radiological conditions in the vicinity of the HSM. These results indicate that the elevated temperature at the surface of the HSM walls may cause cracking or spalling of the walls but that the associated damage will not penetrate the concrete wall to an extent that impacts the ability to meet the established criterion. Although cracking and spalling occur; due to the relatively low thermal conductivity of the concrete and thus the slow movement of the high temperature region through the concrete, it is not felt to progress quickly such that the evaluations regarding depth of spalling are invalidated.

The evaluation of the resulting conditions indicate that:

- concrete temperatures will peak at approximately 1475°F on the surface
- the concrete will exceed ACI 349 limits (350°F) to a depth of only 4.5" into the concrete wall and exceed 1200°F only to a depth of approximately 1" (concrete will maintain 25% to 75% of its compressive strength to a temperature of 1200°F).
- the HSM will be able to function after the fire incident and be in a condition which facilitates HSM repair
- fuel cladding temperatures will be maintained within the fuel cladding short term temperature limit (SAR Table 8.1-13)

- the effect on concrete structural load capacity is minimal since the impact on the concrete is limited to a reduction of load capacity in only 4.5" of the 36" concrete wall thickness

- DSC internal pressure accident limits (50 psig) are maintained

- radiological doses are not significantly increased due to the potential concrete spalling which may occur to a depth of 1".

These above points are based on the evaluations provided in Section 4.2 & 4.3 of calculation CA03945, (Transnuclear West Calculation BGE-01-410). Also, in the event of cracking and spalling to a depth of as much as 4.5", the reduction in shielding would result in an increase in dose rate on the order of a factor of 3. This increase is applied to a dose rate of 7 mrem/hr for the HSM wall surface (SAR Figure 7.4-3. This is not considered a "significant increase in occupational dose" as it involves an increase of no more than a factor of 3 and a total task estimate of dose to repair of less than 100 man-mrem. In addition to the short duration of repair actions, dose rates to the public, on or beyond the nearest boundary due to the fire accident, would be negligible, and would not result in an increase beyond the limits of 10CFR72.106 (5 Rem). A revision to SAR section 8.2.10 is required for this analysis revision and a markup is attached.

USAR Revision No: - 8
USAR Sections Reviewed - 3.2, 8.2
Tech Spec Bases Amendment/Rev No: - 1
Tech Spec Bases Reviewed: - B 2.1

Section 3.2 describes the Structural and mechanical Design Criteria

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

Probability of Malfunction

The analysis revision does not alter the DSC or the interior of the HSM configuration and will therefore not alter the probability of a malfunction of the system or its components. The only component affected is passive and continues to perform its function after the forest fire. The change does not affect the structural integrity of the HSM. The change does not impact system loading and unloading operations

Consequences of Malfunction

The analysis revision does not alter the DSC in a manner that will alter the consequences of a malfunction. The HSM interior and DSC temperatures are affected minimally and dose rates to the public are unaffected. Only the surface dose is increased to a maximum of 21 mrem/hr from 7 mrem/hr which only affects occupational doses. The revised analysis is not associated with system malfunctions or the evaluations of the consequences of system malfunctions. The change does not impact system loading and unloading operations.

Probability of Accident

The analysis revision does not alter the DSC or HSM configuration or their interaction with the environment in any way that could impact the probability of occurrence of an accident.

Consequences of an Accident

The increased HSM temperature due to the revision of the forest fire accident analysis does not result in an increase in the consequences of this accident; there are no consequences. The HSM interior and DSC temperatures are affected minimally and dose rates to the public are unaffected. Only the surface dose is increased to a maximum of 21 mrem/hr from 7 mrem/hr which only affects occupational doses.

2 The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created

Possibility of New Malfunction

The possibility of a new or different malfunction is not created since the forest fire scenario is already addressed in the SAR. The discussion above documents the acceptability of these accident results on the integrity of the DSC/HSM design requirements.

Possibility of New Accident

The possibility of a new or different accident is not created since the forest fire scenario is already addressed in the SAR. The discussion above documents the acceptability of these accident results on the integrity of the DSC/HSM design requirements.

3. **The margin of safety as defined in the basis for any Technical Specification is not reduced**

The margin of safety as defined in the basis for the BGE ISFSI Technical Specification 2.4 is not reduced since the analysis revision documents that the stress and dose allowables specified in the SAR and SER are still maintained (see above for detailed discussion). T/S 2.4 requirements for HSM dose rates are associated with initially loaded HSMs. The actions associated with post fire repair are considered to be within the Action required by the T/S

Thus, there is no Unreviewed Safety Question

Occupational Dose

There is not a significant increase in the occupational dose as discussed above. The normal maximum dose rate is 7 mrem/hr and is applicable to initially loaded HSMs. The forest fire is estimated to result in a maximum increase by a factor of 3. This is not considered a significant increase in occupational dose especially considering the short duration of repair actions

Environmental Impact

There is no environmental impact due to the effect of the forest fire on the HSM

Summary: (For NRC Annual Report, provide a brief overview)

This activity involves a change in the ISFSI SAR in order to consider a revision of the Forest Fire Accident analysis which resulted in an increased HSM concrete temperature. The temperature exceeds the ACI 349 limits of 350F to a depth of 4.5" although 25% to 75% of its strength is retained up to 1200F which occurs to a depth of 1". Some spalling and cracking may result but the structural integrity is not impaired and DSC temperatures are only minimally affected. Also, dose rate at the surface is increased but there is not a significant increase in the occupational dose and the dose to the public is not affected.

ATTACHMENT 3, SAFETY EVALUATION FORM

Page 1 of ~~8~~ ~~7~~ 8

ACTIVITY: **UCR 00104** **50.59 Log No.:** **72.48 Log No.:** **SE00153**

Removal of Q&A from Appendix A of the ISFSI USAR

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES X NO Involve an unreviewed safety question (USQ)?
☐ YES X NO Involve a change in the Technical Specifications/License Conditions?
X YES ☐ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES X NO Involve a Significant Increase in Occupational Dose?
☐ YES X NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Todd Baumbach J.E. REMENIUK Todd Baumbach/JES
PRINTED NAME AND SIGNATURE
Department: Sargent & Lundy
Date: 8/8/00

X YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: Bob Beall **PRINTED NAME**	Resp. Ind.: Chris Dobry **PRINTED NAME**	Resp. Ind.: Getachew Tesfaye **PRINTED NAME**
SIGNATURE	**SIGNATURE**	Getachew Tesfaye **SIGNATURE**
Work Group: **NFM**	Work Group: **PES**	Work Group: **NRM**
Date: 10/13/00	Date: 10/12/00	Date: 10/18/00

Approved ☒ Disapproved ☐
Signature Vic Suchodolski
INDEPENDENT REVIEWER
Date 9/8/2000

Approved ☑ Disapproved ☐
Signature
W.E. KEMPER **GS-DES, GS-TSES, or PE-PDSU**
Date 10.20.00

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 00-092 Date: 10/25/00

Recommend Approval ☑ Recommend Disapproval ☐ Signature **POSRC CHAIRMAN** Date 10/25/00

Approved ☑ Disapproved ☐ Signature **PLANT GENERAL MANAGER** Date 10/26/00

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ YES ☑ NO
Signature: W Russell Date: 3/7/01
OSSRC SES Chairman
If yes, OSSRC Meeting No.

Proposed Activity:

This activity (UCR 00104) involves the removal of questions/responses from Appendix A of the Independent Spent Fuel Storage Installation (ISFSI) Updated Safety Analysis Report (USAR). The questions being deleted, and the basis for deletion, are shown on Attachment 1 of this evaluation. The questions/responses contained in the ISFSI USAR, Appendix A are the formal NRC questions and the BGE responses that were generated during the initial licensing of the ISFSI. The majority of the responses have been incorporated into the USAR text. However, many of the questions/responses (see Attachment 1) go beyond the level of detail required in the USAR and will be deleted.

NUREG-1536, Final Report, published January 1997, "Standard Review Plan for Dry Cask Storage Systems" was reviewed and verified that no USAR information deleted was described in the NUREG-1536.

This activity is supported by NEI 98-03, Revision 1 "Guidelines for Updating Final Safety Analysis Reports", which has received NRC endorsement per Regulatory Guide 1.181. Specifically, Appendix A, Page 4 of NEI 98-03, Revision 1 states:

"The following types of excessively detailed textural information may be removed from UFSARs, except as indicated by applicable regulatory guidance or NRC Safety Evaluation Reports:

> *Criterion 1 – Descriptive information that is not important to providing an understanding of the plant's design and operation from either a general or system functional perspective, e.g., component model numbers*
>
> *Criterion 2 – Design information that is not important to the description of the facility or presentation of its safety analysis and design basis, e.g., component details such as specific motor horsepower ratings for MOVs*
>
> *Criterion 3 – Design information that, if changed during the life of the plant, would have no impact on the ability of plant systems, structures, and components described in the UFSAR to perform their design basis function(s), e.g., specific HVAC equipment capacity and flow rate information for structures that do not contain equipment that performs design basis functions*
>
> *Criterion 4 – Analytical information, e.g., detailed calculations, that is not important to providing an understanding of the safety analysis methodology, input assumptions and results, and/or compliance with relevant regulatory and industry standards."*

NOTE: The last column on Attachment 1 identifies which of the above criteria is used as a basis for deleting the associated question/response from Appendix A of the ISFSI USAR.

An NEI document titled, "NEI 98-03, Revision 1, Guidelines for Updated FSARs", signed by Anthony R. Pietrangelo, NEI Licensing Director, dated June 30,1999 was distributed to utilities. This document consists of a series of Utility questions and NEI answers relating to specific implementation of NEI 98-03. Specific to removal / maintenance of question / response issues, Question 11 from "NEI 98-03, Revision 1, Guidelines for Updated FSARs" inquired about the status of formal NRC questions and responses with respect to inclusion in the UFSAR. The NEI response states as follows:

> *"The Q & A that were submitted to the NRC during review of the initial license application remain in the docket file. Per the Questions and Responses (Q & R) Concerning the Updated Rule provided in GL 80-110 and GL 81-06, the responses should have been appropriately incorporated in the "body" of the updated FSAR. It would be expected that the level of detail used when including the responses in the FSAR would not exceed the level of detail for information that was typically included in the FSAR at that time. Some responses may not warrant incorporation in the FSAR at all, e.g., where a response provided additional information to justify the adequacy of the FSAR as written. While licensing practices may vary with respect to consideration of the Q & R as part of 10 CFR 50.59 evaluations, the Q & R are not considered part of the UFSAR and are therefore not within the scope of 10 CFR 50.59. To the extent responses are incorporated in the body of the UFSAR or as a separate volume, the information is subject to 10 CFR 50.59.*

> *Commitments made in the responses, regardless of whether they were incorporated in the FSAR, remain commitments in the docket file unless the licensee has taken appropriate actions to revise or remove them.*
>
> *Because the Q&R are not considered part of the UFSAR, they are not "historical information" as that term is used in NEI 98-03. Q&R information incorporated into the body of the FSAR would be historical only if it meets guidelines for historical information provided in NEI-98-03."*

The inclusion of the NRC questions and BGE responses into the USAR was done as a matter of convenience by BGE, and was not required by the NRC.

Relevant historical examples of applications for Appendix A in the ISFSI USAR include:

- The BGE response on USAR, Page A.3-7. The response stated, "Since the governing load combinations are not affected by this change, the SAR need not be revised." It is indicative from this response that the NRC did not consider the BGE responses part of the SAR text.
- In other instances, BGE stated that the SAR would be revised based on the NRC questions. For example, the response to question DSC-2 (USAR, Page A.C-1) states that SAR Tables 8.2-9 and 8.2-10 would be revised.

Reason for Activity:

Many of the questions/responses contained in Appendix A of the USAR contain excessive detail that goes beyond that typically required in the USAR. Additionally, engineers reviewing the USAR for 72.48 safety evaluations must look in more than one location (the text and Appendix A) to verify that the activity does not result in a change to the USAR. Incorporating the responses into the text and eliminating the ones whose detail exceeds that required of the USAR will make the task of reviewing the USAR easier.

Function(s) of affected SSCs:

NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. There are three major components of the NUHOMS-24P system. Those three components are: 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); and 3) Horizontal Storage Module (HSM).

NUHOMS-24P – the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules are being built incrementally, as needed, to match BGE's requirements for additional storage. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for storage.

Dry Shielded Canister (DSC) – the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Task Cask (TC) – the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunions near the top of the cask for down ending / up righting and lifting of the cask in the Auxiliary Building. The two lower trunions service as the axis of rotation during down ending / up righting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

Horizontal Storage Module (HSM) – each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure that consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two-foot thick interior walls that separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide temporary storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

Criticality Control – The DSC design provides for sub-criticality during the wet loading, DSC drying, and interim storage operations. This is accomplished by a combination of mechanical separation of the fuel assemblies by the internal basket assembly and neutron absorption in the steel guide sleeve material.

Fuel Support and Configuration Control – The DSC internal basket assembly provides support for the spent fuel assemblies during normal operations. The DSC also provides support for the spent fuel assemblies during normal operations. The DSC also provides configuration control related to post accident recovery of spent nuclear fuel. The DSC is designed so that the worst-case postulated accidents, including a cask drop, will not result in deformation of the Internal Basket Assembly or the DSC shell to such a degree that retrieval of intact fuel assemblies is not assured. The structural characteristics of the Transfer Cask (TC) and the DSC limit the deceleration loads on the fuel assemblies so that their integrity is assured in the worst-case drop accident.

Shielding – The DSC materials provide gamma radiation shielding. The DSC provides gamma shielding at its ends by the use of lead shield plugs. These provide ALARA dose rates at the top of the canister during drying and sealing operations and at the bottom for minimizing dose rates during DSC loading into the Horizontal Storage Module (HSM) and at the HSM door during storage. The shielding function is achieved by the outer canister portion of the DSC.

Thermal – Decay heat is removed by thermal radiation and conduction from the DSC to the TC, and the thermal radiation and conduction and convection from the DSC and the HSM. The DSC maintains the helium cover gas, which is required for corrosion control. This cover gas improves the thermal performance of the DSC. The decay heat removal function is achieved by the outer canister portion of the DSC.

SAR Revision No.: 9
SAR Sections Reviewed: All, including Appendix A
Tech Spec Bases Amendment/Rev No.: 2
Tech Spec Bases Reviewed: 2 and 3/4

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ Yes X No

May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: The activity involves the deletion of several questions/responses (See Attachment 1) from Appendix A of the ISFSI USAR. The drawings and calculations referred to in the questions/responses are controlled under a BGE engineering document control procedures. Changes to these documents are subject to independent verification and supervisory approval. Information important to safety is not deleted. As shown in Attachment 1, each of the deletion meets one of the criteria of NEI 98-03 for the acceptable deletion of information from the USAR. The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased due to this administrative activity. No physical modification to the facility is being made due to this activity. This activity does not degrade the reliability of components required to support the plant safety functions since the design of the facility is unchanged.

☐ Yes X No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: The BGE responses that are being deleted made changes to the SAR text and other assorted documents (drawings, calculations, etc.) when required to satisfy NRC requirements/requests. In many instances, the BGE response simply justified the existing design. The activity is administrative in nature and does not result in a physical change to the ISFSI facility or the manner in which it is operated or maintained. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR are not increased by this activity.

☐ Yes X No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: Credible accidents analyzed for the ISFSI facility are described in Section 8.2 of the USAR. Any changes to the controlled documents associated with the questions/responses being deleted are subject to independent review and are controlled under BGE engineering document control procedures. Information important to safety is not deleted in this activity. As shown in Attachment 1, each of the deletion meets one of the criteria of NEI 98-03 for the acceptable deletion of information from the USAR. Therefore, the probability of an accident previously evaluated in the SAR will not be increased.

☐ Yes X No May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident: Because this activity is administrative in nature and does not result in a physical change or modification to the ISFSI facility, the consequences of an accident, including radiological dose consequences, previously evaluated in the SAR will not be increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ Yes X No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction: No new equipment, procedures, or tests are being added to the ISFSI facility as a result of this activity. No physical changes are being made to existing ISFSI equipment. Information important to safety is not deleted. As shown in Attachment 1, each of the deletion meets one of the criteria of NEI 98-03 for the acceptable deletion of information from the USAR. Therefore, the possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created.

☐ Yes X No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident: This activity is administrative in nature and does not result in any physical changes to the ISFSI facility. No new equipment, procedures, or tests will result from this change. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR will not be created.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ Yes X No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
2.3	The deletion of questions / responses pertaining to the transfer cask drop analysis (Bases 2.3) will not result in a reduction in the safety margin of the Technical Specifications. No physical change to the facility is being made due to this activity. The changes are administrative in nature. Any changes to the calculations associated with these questions / responses are controlled under BGE engineering document control procedures and processes that require independent verification and supervisor approval for revision.
2.4	The deletion of questions / responses pertaining to the horizontal storage module dose rates (bases 2.4) will not result in a reduction in the safety margin of the Technical Specifications. No physical change to the facility is being made due to this activity. The changes are administrative in nature. Any changes to the calculations associated with these questions / responses are controlled under BGE engineering document control procedures and processes that require independent verification and supervisor approval for revision.
3 / 4.5	The deletion of questions / responses pertaining to the forest fire analysis (Bases 3 / 4.5) will not result in a reduction in the safety margin of the Technical Specifications. No physical change to the facility is being made due to this activity. The changes are administrative in nature. Any changes to the calculations associated with these questions / responses are controlled under BGE engineering document control procedures and processes that require independent verification and supervisor approval for revision.

Complete for 72.48:

☐ Yes X No Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose: The changes associated with this activity are administrative in nature and will result in no physical changes to the ISFSI facility. No new tests, experiments, or procedures are generated as a result of this activity. Therefore, the proposed activity will not result in an increase in occupational dose.

☐ Yes X No Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact: No physical changes are made to the ISFSI facility. No changes are being made to the manner in which the ISFSI facility is being operated or maintained. Therefore, this administrative activity will not result in an unreviewed environmental impact.

ACTIVITY:	UCR 00104	50.59 Log No.:		72.48 Log No.:	SE00153

Summary: (For NRC Report, provide a brief overview)

This activity deletes questions / responses from Appendix A of the ISFSI USAR. No physical changes to the ISFSI facility are being made by this proposed activity. No new tests, experiments, or procedures result from his activity. The information being deleted by this activity is not required per the guidance of NUREG-1536 (SRP for Dry Cask Storage Systems). It constitutes excessive detail as defined in NEI 98-03, revision 1, "Guidelines for Updating Final Safety Analysis Reports".

USQ Determination: This activity was evaluated against the criteria of 10CFR72.48(a)(2), such as the probability of occurrence or the consequences of an accident or the malfunction of equipment important to safety, and it was concluded that it does not involve an unreviewed safety question (USQ).

Attachment 1

Page	Question	Basis for Deletion	Criterion
A.3-1 through A.3-7	N/A	The question/response pertains to stress analysis calculations that are beyond the level of detail required in the SAR. The BGE Response simply justified previously submitted information.	4
A.7-1	7.0-1	The question was simply asking the current status of BGE administrative processes with respect to the ISFSI. The Performance Improvement Plan no longer exists.	3
A.8-1 through A.8-4	8.0-9 (Additional Information) SAR Table 8.2-8	The questions/responses are centered on the forest fire analysis (calculation), or the specific details of how spacer disc stresses were calculated (BGE calculation 001.020), that go beyond the level of detail required in the SAR. The forest fire calculation was revised in 1998 and the USAR will be revised via approved 10CFR72.48 safety evaluation SE000152.	4
A.A-1 through A.A-6	Drawing-2, -3, -5 through –8, -10 through –13, 085-1, 087-1	The BGE response referred to revised drawings. No changes were required.	1
A.C-1 through A.C-3	DSC-1, -2, -4, -6 through -8	The question/response is associated with a calculation that goes beyond the level of detail required in the SAR. Changes to conform to NRC request/requirements were made, where required.	4
A.D-1	DSC-SUPT-1	The BGE response simply provided a revised calculation.	1
A.D-1	DSC-SUPT-3	The question/response is associated with a calculation that goes beyond the level of detail required in the SAR. No changes to the calculation were required.	4
A.D-1	DSC-SUPT-4	The question/response is associated with a calculation that goes beyond the level of detail required in the SAR. The BGE response simply justified the existing design.	1
A.D-2 through A.D-4	DSC-SUPT-5	The question/response is associated with a calculation that goes beyond the level of detail required in the SAR. The BGE response simply justified the existing design. No changes to the calculation were required.	1
A.F-1	HSM-1	The question/response is associated with a calculation that goes beyond the level of detail required in the SAR. The BGE response simply justified the existing design. No changes to the calculation were required.	4
A.F-2 through A.F-9	HSM-4	The question/response is associated with a calculation that goes beyond the level of detail required in the SAR. SAR changes to conform to NRC requests/requirements were made, where required.	4
A.F-10	HSM-5	This is an editorial comment, as noted in the NRC question.	1
A.G-1	TC-4	The question/response is associated with a calculation that goes beyond the level of detail required in the SAR. The BGE response simply justified the existing design. No changes to the calculation were required.	4
A.H-1 through A.H-3	YOKE-1	The question/response is associated with a calculation that goes beyond the level of detail required in the SAR. Changes to conform to NRC requests/requirements were made, where required.	4

ATTACHMENT 3, 10 CFR 50.59/10 CFR 72.48 EVALUATION FORM (Page 1 of 13)

PROPOSED ACTIVITY NO: ES199800410

50.59 Log No.: N/A

72.48 Log No.: SE00154

Based on this 10 CFR 50.59/10 CFR 72.48 evaluation, does the Proposed Activity:

☒ YES ☐ NO Require a License Amendment for a change to the Technical Specifications/License Conditions?

☒ YES ☐ NO Require a License Amendment because it meets one (or more) of the eight (8) criteria of 10 CFR 50.59(c)(2)/10 CFR 72.48(c)(2)?

Prepared by:	B. H. Scott/ M. Kaiseruddin	Department:	MEU/Sargent & Lundy	Date:	7/9/01
	PRINTED NAME AND SIGNATURE				

☒ YES ☐ NO Are cross-disciplinary concurrence reviews needed?

If 'YES', document completion of these reviews below:

Responsible Individual		Responsible Individual		Responsible Individual	
G. Tesfaye		~~J. E. Remeniuk~~		R. H. Beall	
PRINT NAME		PRINT NAME		PRINT NAME	
Licensing		~~Project Management~~		NFM	
PRINT ORGANIZATION		PRINT ORGANIZATION		PRINT ORGANIZATION	
	07/10/01		/	R H. Beall	7-9-01
SIGNATURE	DATE	SIGNATURE	DATE	SIGNATURE	DATE

Print/Signature:	J. R. Sponsel	Print/Signature:	CJ LUDLOW
	INDEPENDENT REVIEWER		GS-DES, GS-TSES, or PE-PDSU
Date:	7/12/01	Date:	7/12/01

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 01-055 Date: 7/18/01

Recommend Approval ☑ Recommend Disapproval ☐ POSRC CHAIRMAN Date 7/18/01

Approved ☑ Disapproved ☐ PLANT GENERAL MANAGER Date 7/18/01

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? ☐ YES ☐ NO N/A

Print/Signature: W T Russell Date: 9/11/01

OSSRC SES Chairman

If "YES", OSSRC Meeting No. ______

LAR reviewed and approved at OSSRC MTG 01-04 on July 26, 2001.

PROPOSED ACTIVITY NO: ES199800410

50.59 Log No.: N/A

72.48 Log No.: SE00154

Proposed Activity (Description):

The proposed activity consists of making changes to the Independent Spent Fuel Storage Installation (ISFSI) USAR [Refs. 1 and 2]. The changes are based upon six new analyses performed to document the fuel assembly structural integrity under Transfer Cask drop scenarios. The analyses are documented in Constellation Nuclear / Calvert Cliffs Nuclear Power Plant (CCNPP) calculations CA04678, CA04679, CA04680, CA05673, CA05797 and CA05760 [Refs. 3 through 5, 14, 18, and 19].

The proposed activity addresses the fuel assemblies utilized in Units 1 and 2, batches A through J. The above mentioned calculations are based on a fuel assembly mass of 1450 lbs., which is conservative compared to the Tech Spec limit of 1300 lbs. In compliance with the Tech Sec limit, only those fuel assemblies with masses less than 1300 lbs. will be loaded into the ISFSI.

The changes to the USAR consist of the following:

1. Section 3.3.4.1, second paragraph, is being revised to replace the phrase "the DSC basket is designed to maintain the fuel configuration after a drop accident", with the phrase "the DSC basket is designed to keep the fuel assemblies separated from each other even after a drop accident".
2. Section 4.2.3.2, fifth paragraph, is being revised to delete the following sentences, "Additionally, the structural characteristics of the cask and DSC limit the deceleration loads on the fuel assemblies so that their integrity is assured in the worst case drop accident (Reference 4.4). Thus, retrievability of fuel from the ISFSI and from the DSC is assured, even following the maximum credible accident".
3. Section 5.1.1.9 is being revised to state that the DSC, transfer cask, and fuel shall be examined following any accidental drop.
4. Section 8.2.5.2 is being revised to replace the last sentence about fuel integrity with several paragraphs, which discuss the integrity of each of the components of the fuel assembly based on the new analyses.
5. Section 8.4 is being revised to add new references 8.33 through 8.39.

A review of the Technical Specifications has revealed that the following changes need to be made in them, based on the proposed activity.

1. Section 2.3: The ACTION statement will be revised to delete the phrase "from a height greater than 15 inches, (0.38 m)".

In addition, the basis for the above Technical Specification Section will be revised to delete the last sentence, which is related to the 15" drop height.

Background

The Independent Spent Fuel Storage Installation (ISFSI) at the Calvert Cliffs Nuclear Power Plant (CCNPP) stores spent fuel assemblies within Dry Shielded Canisters (DSCs). Twenty-four spent fuel assemblies are loaded into each DSC. Each DSC contains an outer leak-tight shell and an internal basket assembly. The outer shell provides the structural strength, shielding, and a leak-tight chamber for containing helium. The internal basket assembly includes twenty-four stainless steel guide sleeves (one for each spent fuel assembly), nine perforated carbon or stainless steel spacer discs,

PROPOSED ACTIVITY NO: **ES199800410** **50.59 Log No.:** **N/A**

72.48 Log No.: **SE00154**

and four carbon or stainless steel support rods. The nine spacer discs are spaced out along the length of the DSC at locations that approximately coincide with the spent fuel assembly's eight spacer grids and the single lower retention grid. The spacer discs are not structurally attached to the DSC shell walls or inner cover plates. The guide sleeves traverse the length of the DSC cavity through openings in the nine spacer discs. Four support rods are used to maintain the spacer disc locations. The support rods traverse the length of the DSC cavity through the spacer discs, and are structurally welded to the spacer discs. The DSC is loaded into a Transfer Cask (TC) for transporting it to the storage facility, where it is placed in a Horizontal Storage Module (HSM) for a long-term on-site storage.

While an accidental drop of the TC is not considered credible, it is still postulated to occur, and its impact on the integrity of the DSC and TC is analyzed in detail and documented in the USAR. The impact of the drop on the integrity of the fuel assemblies was not analyzed, but was inferred from a Lawrence Livermore National Laboratory Report [Ref. 7].

As part of the 1998 Calvert Cliffs Nuclear Power plant "Independent Spent Fuel Storage Installation (ISFSI) Loading Restart Recovery Project," CCNPP performed an extensive evaluation of the documentation available for the Dry Shielded Canister (DSC) and its internals, to ensure that all its licensing requirements are satisfied. During the evaluation, CCNPP discovered that the spent fuel assembly safety, during a TC/DSC accident drop scenario, was not adequately demonstrated in Ref. 7, which was widely used by industry as it was distributed by the NRC. A review, by CCNPP, of the Lawrence Livermore report revealed that only the cladding was evaluated for structural adequacy. Other components of the CCNPP spent fuel assembly were not evaluated in the Lawrence Livermore report. The failure of these components could compromise the integrity of the fuel rod cladding, affect the center-to-center distance between fuel rods which could impact the criticality and thermal evaluations, and could hinder the retrievability of the spent fuel assembly from the DSC. Further, the latest NRC guidance [Ref. 8] has raised questions about the method and assumptions used in the analysis of the fuel rods. New analyses were prepared to address these issues.

The Westinghouse/CE 14X14 fuel assemblies, utilized by CCNPP, are approximately 157 inches in length and approximately eight by eight inches square in cross section. Each fuel assembly contains one hundred and seventy-six, 0.44 inch diameter fuel rods, which are closely bundled together. The fuel rods are held in place by an assembly of eight spacer grids, one retention grid, five guide tubes, an upper end fitting, and a lower end fitting. Each fuel rod contains spent nuclear fuel pellets, encased in a Zircalloy cladding. The cladding performs the safety function of preventing the release of fission product. Twenty-four spent fuel assemblies are loaded into a DSC. Within the DSC, the fuel assemblies are loosely held in place by guide sleeves belonging to the internal basket assembly.

New Analyses

New analyses were prepared to determine the response of fuel assembly components to a postulated TC/DSC drop. Reference 3 was prepared first to evaluate the response of all of the fuel assembly components. Subsequently, References 4 and 5 were prepared to

PROPOSED ACTIVITY NO: ES199800410 **50.59 Log No.:** N/A

72.48 Log No.: SE00154

perform more detailed evaluations of the spacer grid and the upper end fitting (UEF), and therefore, supersede specific portions of Reference 3 associated with these components. Reference 3 documents the responses of fuel rods, guide tubes, lower end fitting (LEF) and the retention grid. In these analyses, the DSC was not credited with providing any reduction in the deceleration of fuel assemblies during the drop.

The mass of the fuel assembly used in these analyses was such that it enveloped different types of fuel. Reference 6 provides the variations in fuel assembly masses. The maximum mass is seen from the Reference as 1360 lbs., to which 80 lbs. should be added to address the possibility of having control components included in an assembly. Therefore, the maximum fuel assembly mass was conservatively taken as 1450 lbs. It is noted, however, that CCNPP is currently limited by its Tech Specs to load into the ISFSI only those fuel assemblies whose masses are less than 1300 lbs. each.

The temperatures at which the ASME code allowables were determined in all of the analyses envelop the maximum cladding temperature of 635 ^{O}F, per Technical Specification 3.4.1.

The objectives of Reference 3 were to evaluate the fuel assembly retrievability and criticality following a drop. Fuel assemblies from Units 1 and 2 batches A-J were included in the evaluation. Axial drops, both right side up and upside down, and lateral drop were analyzed for the licensing basis deceleration of 75g. The oblique drop was not analyzed, however, the design basis deceleration for the oblique drop is bounded by the design basis decelerations of axial and lateral drops. The axial drop analysis was based on the results of a similar analysis reported in the Electric Power Research Institute (EPRI) report, EPRI NP-7419 [Ref. 9]. The lateral drop analysis was performed using finite element methodology, through which Von Mises stress intensities were calculated and compared against the allowables. Selected results of Reference 3 were updated by Reference 20 for a minor change in one of the parameters.

The calculated stress intensities and the allowables, in units of ksi, are tabulated below.

	Lateral	Axial (Right Side Up)	Axial (Upside Down)	Allowable
Fuel Rods	46.84*	30.3 (for a weaker 17X17 fuel at 88g)	<30.3	58.95
Guide Tubes	51.43*	44.59	44.59	57.96*
LEF & Retention Grid	4.075	36.45	Insignificant	57.15

*Taken from Reference [20].

It is seen that the calculated stresses are less than the allowables, therefore, no failures would occur.

Reference 4 analyzed the response of UEF flow plate, based on the consideration of the elastic-plastic material behavior. The analysis considered a top-end vertical drop scenario, which is the most limiting scenario for the UEF. The acceptance criterion used was the lack of failure, rather than meeting the code minimum stress and strain values. It was determined that after a 75g drop the UEF ligaments would not fail, and that at

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least a factor of 2 margin would be available against ductile tearing.

Reference 5 analyzed the structural integrity of spacer grids, in order to demonstrate fuel retrievability and cladding integrity. The guide sleeve, in which the assembly resides, was assumed to be perfectly rigid for conservatism. An equivalent static impact force of 75g was used in the analysis to represent the worst case loading experienced by the spacer grids during a DSC accidental horizontal drop. The Zircaloy-4 spacer grid, as a part of a spent fuel assembly is expected to be irradiated and thus exhibit brittle behavior. Nonetheless, both brittle and ductile (un-irradiated) cases were considered in the calculation.

Based on the results of this evaluation, it was determined that major structural damage of the spacer grid would occur from a 75g accident drop scenario for both ductile and brittle material behavior assumptions. The evaluation also showed that some damage would occur to the spacer grids even for drop heights of less than 15 inches, for both ductile and brittle material behavior assumptions. Therefore, ISFSI Technical Specification 2.3 criterion of a drop height of 15 inches or more for the fuel assemblies to be inspected for damage is incorrect. Also, the ISFSI USAR assertion, of a reasonable assurance that no damage would occur for drop heights of less than 15 inches, is incorrect. In addition, the perimeter strip would likely fail at the lower levels of the spacer grid and thus allow fuel rods to be relocated from their original grid locations, and create the possibility of one of them getting wedged against the guide sleeve. It was further determined that, with this possibility, an additional pull force of about 220 lbs. would be required for retrieving the fuel assembly from the DSC. This additional required pull force is less than the pull force required in case of clip angle failure scenario [Ref. 12], for which the fuel assembly retrievability was previously determined to be feasible. Therefore, retrievability of the fuel assembly from the DSC would not be compromised.

The effect of impact of a broken spacer grid fragment, during a horizontal drop, on the fuel rod cladding was investigated [Ref. 18]. Various cladding vs. fragment orientations and edge conditions were considered. The maximum cladding wall stress was found to be less than the allowable stress of 80.5 ksi.

The failure of spacer grids was determined also to cause a change in the fuel rod pitch from 0.58 inch to 0.465 inch. The impact of this pitch reduction was evaluated in Reference 14. The cases evaluated were: an optimum density helium moderated and 0, 25, 75 and 100% collapsed localized assembly, an unborated fully moderated 0, 25, 75 and 100% collapsed localized assembly, and an unborated fully moderated 0, 25, 75 and 100% collapsed DSC system. In all of the above cases the k_{eff} was calculated to be below the regulatory limit of 0.95.

Reference 19 evaluated the cladding temperature for the reduced rod pitch, and determined that it would be lower than that for the normal rod pitch because of better conductivity of the new arrangement.

References:

1. Calvert Cliffs Independent Spent Fuel Storage Installation USAR, Rev. 9
2. CCNPP ISFSI USAR Change Request, UCR-00180

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	3. CCNPP Calculation CA04678, Rev. 0000, BGE (Calvert Cliffs Units 1 & 2) ISFSI Dry Storage Cask Drop Analysis 4. CCNPP Calculation CA04679, Rev. 0000, ISFSI Fuel Assembly Upper End Fitting Structural Integrity 5. CCNPP Calculation CA04680, Rev. 0000, ISFSI CE 14X14 Fuel Grid Horizontal Drop Evaluation 6. BGE Memorandum NEU 99-164, from T. A. Schearer to G. V. Patel, et al., 7/7/1999 7. Lawrence Livermore National Laboratory Report, UCID-21246, Dynamic Impact Effects on Spent Fuel Assemblies, 10/20/1987 8. NRC Interim Staff Guidance ISG-12, Rev. 1, Buckling of Irradiated Fuel Under Drop Conditions 9. Electric Power Research Institute Report, EPRI NP-7419, Fuel-Assembly Behavior Under Dynamic Impact Loads Due to Dry-Storage Cask Mishandling, July 1991 10. CCNPP Issue Report IR3-007-608, 03/04/1998 11. Technical Specifications for Calvert Cliffs ISFSI, Amendment 3 12. CCNPP Calculation CA04132, Rev. 0003, Nutech Horizontal Module System (NUHOMS) 24P ISFSI Dry Shielded Canister Structural Analysis for DSC Numbers R001-R024 13. CCNPP Calculation CA04141, Rev. 0002, ISFSI Transfer Cask Structural Analysis 14. CCNPP Calculation CA05673, Rev. 0000, Criticality Analysis for Dropped Fuel Storage Cask 15. NEI 96-07, Rev. 1, Guidelines for 10 CFR 50.59 Safety Evaluations, 11/00 16. NRC Interim Staff Guidance ISG-3, Rev. 0, Post Accident Recovery 17. Calvert Cliffs ISFSI Updated Environmental Report, Rev. 1 18. CCNPP Calculation CA05797, Rev. 0000, DSC Horizontal Drop – Fuel Rod Cladding Integrity during Impact with a Broken Spacer Grid Fragment 19. CCNPP Calculation CA05760, Rev. 0000, Effective Conductivity of the Reconfigured Fuel Assembly 20. Hopper and Associates Letter HABGE-02/01-0952, ITR Issue Resolution – Fuel Rods and Guide Tubes, 02/13/01
Reason for Activity:	This activity is being carried out in part to resolve Issue Report IR3-007-608 [Ref. 10]. The report was written to identify the issue related to the fuel assembly integrity. The ISFSI USAR claimed, based on the Lawrence Livermore Report [Ref. 7], that the fuel assembly integrity was maintained during and following a cask drop. A review of the report showed that it evaluated only the cladding, but not any of the other fuel assembly components. The design objectives of the dry storage system include ensuring that the retrieval of fuel assemblies is still assured and that the fuel rod integrity is not compromised, following the worst case postulated cask drop accident, ISFSI Technical Specifications Section 2.3, "Transfer Cask", states that the transfer cask lifting height with a non-single-failure-proof lifting device shall not exceed 80 inches. The Technical Specification also states that for drops greater than 15 inches the DSC will be returned to the spent fuel pool and be visually inspected. Therefore, retrievability of

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	fuel from the DSC is demonstrated through analyses. The stipulation of 15 inches drop height needs to be deleted, so that the inspection of the DSC will be required after a drop from any height. ISFSI Technical Specifications Section 3.4.1 limits the maximum air temperature rise within the Horizontal Storage Module. This is based on limiting the temperature of the hottest rod in the DSC to below 635°F. The potential impact of a reduction in rod pitch on cladding temperature is addressed in this safety evaluation.
Function(s) of affected SSC:	The affected SSCs are the spent fuel assemblies from Units 1 and 2, batches A through J. CCNPP utilizes Westinghouse/CE 14X14 design fuel assemblies. Each fuel assembly contains one hundred and seventy-six fuel rods, which are 0.44-inch diameter each. The fuel rods are held in place by a set of eight spacer grids, one retention grid, five guide tubes, an upper end fitting, and a lower end fitting. Each fuel rod contains spent nuclear fuel pellets, encased in a Zircalloy cladding. The cladding performs the safety function of containing the fission products and preventing their release. The guide tubes, spacer grids, and end fittings form the structural frame of the assembly. The fuel assembly structural components are also effective in limiting bending stresses in fuel rod cladding, and in improving fuel rod stability under axial loads. The spacer grids maintain the fuel rod pitch over the length of the assembly. The grids provide positive side restraint to the fuel rods, but only a frictional restraint axially. The spacer grids are the widest part on a fuel assembly and are welded to all five guide tubes. The four outer guide tubes are mechanically attached to the end fittings. The upper end fitting attaches to the guide tubes to serve as an aligning and lifting device. The fuel rods are held together in the assembly at a pitch of 0.58 inch, which helps to control reactivity. The rods consist of enriched uranium fuel pellets stacked within a Zircalloy cladding tube. The cladding is the first barrier that prevents the fission products from escaping to the outside.

UFSAR Sections reviewed where relevant information was found:	Tech Spec Sections reviewed where relevant information was found:
ISFSI USAR Revision No.: 9 ISFSI USAR Sections Reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key Sections reviewed are listed as follows: 1.2.1 General Description 3.1.1 Materials to Be Stored 3.3.4.1 Control Methods for Prevention of Criticality 4.2.1.2 Dry Shielded Canister (Structural Specifications) 4.2.3.2 Dry Shielded Canister Description	Tech Spec Bases Amendment/Rev No.: 3 Calvert Cliffs Independent Spent Fuel Storage Installation Technical Specifications, Appendix A to Materials License No. SNM-2505, Amendment 3, April 17th, 2001 ISFSI Tech Spec Reviewed: 2.3 Transfer Cask (TC) 3/4.1 Fuel to be Stored at ISFSI 3/4.3.1 Ambient Temperature 3/4.4.1 Maximum Air Temperature Rise 5.0 Design Features

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4.7.3 Individual Unit Description 5.1.1.2 Fuel Loading 5.1.1.9 Removal of Fuel from the Dry Shielded Canister 8.1.1.2 Dry Shielded Canister Analysis 8.1.1.3 Dry Shielded Canister Internal Basket Analysis 8.2.3.2 Earthquake - Accident Analysis 8.2.5 Cask Drop Table 1.2-1 Design Parameters for the Calvert Cliffs ISFSI Table 3.1-1 Principal Design Parameters for Fuel to Be Stored Table 3.3-3 CE 14X14 Fuel Parameters Table 3.3-5 Design Parameters for Criticality Analysis of the DSC Table 3.6-3 Summary of Design Criteria for Accident Conditions Table 8.1-1 Estimated Component Weights Table 8.1-3 Maximum Dry Shielded Canister Stresses for Normal Loads Table 8.1-4 Maximum Dry Shielded Canister Stresses for Off-Normal Loads Table 8.2-1 NUHOMS-24P Accident Loading Identification	

Does the proposed activity:

1. ☐ YES ☒ NO Result in more than a minimal increase in frequency of occurrence of an accident previously evaluated in the UFSAR?

Justification:

Frequency of an Accident:
Accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the USAR. The major accidents consist of loss of shielding, external missiles, earthquake, flood, cask drop, lightning, blockage of air inlets and outlets, DSC leakage, DSC overpressurization, and forest fire. Of these accidents, only the cask drop accident and earthquake incident are impacted by this activity. The earthquake scenario is bounded by the cask drop accident, as the acceleration postulated in a design basis earthquake is 1.5g, which is much smaller than the acceleration in the drop accident of 75g. However, the frequency of occurrence of cask drop,

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or any other accident, is not increased by the new analysis of the fuel assembly integrity.

There is no change to the design or operation of the NUHOMS system caused by this activity. This activity does not modify the external configuration of the DSC envelope. The interface between the DSC and the HSM during ISFSI operations and interim storage of the DSC remains unaffected. Therefore, the frequency of occurrence of an accident involving loss of HSM air outlet shielding, or blockage of HSM air inlets and outlets will not increase.

Pressurization of the DSC due to fuel cladding failure is an accident scenario identified in USAR Section 8.2.9. The limiting DSC pressurization accident event is a rupture of fuel cladding together with blockage of the HSM vents. As stated above, the impact of the cask drop on the cladding was evaluated. It was determined that the fuel cladding would not rupture, and fuel rod integrity would be maintained.

DSC leakage is an accident scenario described in USAR Section 8.2.8. The USAR indicates that there are no credible events that would initiate this type of accident. As stated in the preceding paragraphs, the frequency of an accident that would lead to cladding failure is not increased by this activity. This activity does not affect the design of the DSC pressure boundary and therefore does not increase the probability of DSC leakage.

2. ☐ YES ☒ NO Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component (SSC) important to safety previously evaluated in the UFSAR?

Justification:

Likelihood of Malfunction:

The proposed activity consists of evaluating the spent fuel assemblies contained in a DSC, following a postulated drop, and incorporating the results into the USAR. The analyses were based conservatively on a fuel assembly mass of 1450 lbs., which is higher than the value of 1300 lbs. stated in the USAR and Tech Spec 3.1.1. The analyses showed that the fuel rods, guide tubes, and upper and lower end fittings would maintain their integrity, but the spacer grids would be damaged. A detailed analysis of the damage showed that it would cause the fuel rods to be displaced from their grid location, and potentially get wedged between the assembly and the DSC guide sleeve. However, the analysis also showed that as a result of the wedging the additional force required to retrieve the fuel assembly was estimated to be about 220 lbs. This is less than the pull force required in the previously evaluated scenario of clip angle failure [Ref. 12], for which the fuel assembly retrievability was determined to be feasible. In addition, the fuel rod pitch would reduce from 0.58 inch to 0.465 inch, due to the spacer grid failure. References 14 and 19 established that the decrease in fuel rod pitch would not adversely impact on the criticality control or the cladding temperature.

Therefore, the likelihood of occurrence of a malfunction of equipment important to safety, namely the fuel assemblies, will not be increased.

3. ☐ YES ☒ NO Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR?

Justification:

Consequences of Accident:

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The proposed activity, namely analysis of fuel assembly integrity, is related to the cask drop accident and earthquake incident, as stated above.

The consequences of the cask drop accident on the fuel assemblies were evaluated in References 3 through 5, 14, 18, and 19. The impact on critical safety functions is discussed below. The critical functions affected will be the configuration and criticality controls, and confinement. Other critical functions, such as the shielding, are not affected by the fuel assemblies.

Criticality Control: In the fuel assembly evaluation, the deceleration value of 75g was used for a drop, which ignored any reduction in deceleration provided by the DSC. Reference 5 determined that, following a drop, spacer grids would be damaged, resulting in the reduction of fuel rod pitch to 0.465 inch. Reference 14 evaluated the criticality with reduced rod pitch, and determined that the k_{eff} would still be less than 0.95. Hence, the criticality control is maintained.

Configuration Control: Configuration of fuel assemblies within the DSC needs to be maintained such that the assemblies remain retrievable. During a drop, the spacer grids would be damaged. This could result in a fuel rod being removed from the grid and getting wedged between the fuel assembly and the guide sleeve. It would cause an increase in the extraction force needed to retrieve the fuel assembly. However, the analysis showed that the wedging of the fuel rods would require an additional pull force of only 220 lbs. for fuel assembly removal, which was previously determined to be within the capacity of the spent fuel handling machine. Hence the fuel assembly retrievability is not jeopardized.

Confinement: The drop would not rupture or damage the cladding. Therefore, the radioactive fission products would remain confined within the fuel rods.

Shielding: The fuel assembly, during a drop, would not impact the radiation shielding properties of the DSC, and the DSC shielding materials. Therefore, there is no increase in the probability of a malfunction of the DSC shielding due to this activity.

Thermal Control: The cask drop would result in a reduction of rod pitch. The local cladding temperature was analyzed and determined to be lower than that for the normal rod pitch because of better conductivity of the new arrangement. Therefore, there is no increase in the probability of a malfunction of the DSC thermal control due to this activity.

4. ☐ YES ☒ NO — Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR?

Justification:

Consequences of Malfunction:

The proposed activity determined that the drop accident would not lead to breaching of the cladding, so that consequences of malfunction of equipment important to safety, namely the radiation dose to operators or radiation releases from ISFSI would not increase.

USAR Section 3.3.4.1 states that criticality control is assured by the physical properties and history of the fuel, mechanical control of the assemblies' locations in the DSC basket, neutron absorption by the materials of the basket, Calvert Cliffs administrative controls over fuel identification and handling, and the presence of soluble boron in the fuel pool for wet operations. None of these parameters would change, however, the rod pitch could decrease from 0.58 inch to 0.465 inch. The impact of this reduction of rod pitch on the criticality was evaluated, and found to be insignificant. Therefore, criticality control would be maintained, and there

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would be no increase in the radioactivity content of the fuel assemblies.

5. ☐ YES ☒ NO Create a possibility for an accident of a different type than any previously evaluated in the UFSAR?

Justification:

Possibility of New Accident:

Accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the USAR, and have been discussed previously. Evaluation of the fuel assembly integrity following a drop accident showed that the fuel assembly would maintain its safety functions. The impacts of the drop accident on other components, such as the DSC, and TC were evaluated previously. The results showed that none of the components would fail to perform their safety functions. Since there is no change to the design or operation of the NUHOMS system caused by this activity, the possibility of an accident of a different type than any previously evaluated in the SAR will not be created.

6. ☒ YES ☐ NO Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR?

Justification:

Possibility of New Malfunction:

The proposed activity examines the response of a fuel assembly to a cask drop accident. The evaluation showed that the fuel assemblies would get damaged to certain extent. The components that would get damaged are the spacer grids. Previously, it was inferred from Reference 7 that the fuel assembly would not be damaged. Therefore, the likely failure of fuel assemblies is a malfunction of a different type than any that was previously evaluated in the USAR.

As discussed above, despite the damage the critical functions of the fuel assemblies would remain intact. The cladding would not be breached or ruptured, so that the fission product barrier would be maintained. The criticality would not be a concern, and the thermal response would be within limits. Therefore, the new failure of the fuel assemblies would not cause an adverse impact on safety.

7. ☐ YES ☒ NO Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered?

Justification:

Analyses demonstrated that one component of the fuel assemblies would get damaged following a drop accident, but the fuel assemblies would continue to be able to perform their safety functions of confinement, criticality control, retrievability, and maintaining the temperature limits. The cladding would not be ruptured, thus would be able to maintain the confinement. Fuel rod geometry would be altered, however, analysis showed that it would not compromise the criticality control. Further, analysis showed that the geometry alteration would impact retrievability, but that the assemblies would still be retrievable with an additional pull force of acceptable magnitude. This maintains the original CCNPP commitment regarding retrievability, although the latest NRC guidance [Ref. 16] states, "Recovery methods or the need for Over-Packs or Dry

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Transfer Systems to maintain safe storage conditions would then not be considered and evaluated as part of the licensing process." Since the fuel assembly can still be retrieved after a drop accident, and there is no impact to the integrity of the fuel rods and criticality control, the design basis limit for a fission product barrier is not exceeded or altered.

8. ☐ YES ☒ NO Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses?

Justification:

The impact of an accidental drop of the TC on the integrity of the fuel assemblies was not analyzed previously, but was inferred from a Lawrence Livermore National Laboratory Report [Ref. 7]. Subsequently, CCNPP discovered that the spent fuel assembly safety, during a TC/DSC accident drop scenario, was not adequately demonstrated in Ref. 7. A review, by CCNPP, of the Lawrence Livermore report revealed that only the cladding was evaluated for structural adequacy. Other components of the CCNPP spent fuel assembly were not evaluated in the report. The current activity includes an analysis of the components of the fuel assembly, which were not previously analyzed. The evaluation methods used are the ones that have been used before by CCNPP and the industry. Therefore this activity does not involve a departure from a method of evaluation described in the ISFSI-USAR.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: The proposed activity consists of making changes to the ISFSI USAR, which are based on the new analyses performed to document the fuel assembly integrity under Transfer Cask drop scenarios.

The changes being made to the ISFSI USAR consist of providing a summary of fuel assembly integrity evaluation and results.

It has been determined that the proposed activity will require the following Technical Specifications changes.

Section 2.3: The ACTION statement needs to be revised to delete the phrase "from a height greater than 15 inches, (0.38 m)"

Reason for Activity: This activity is being carried out to resolve an issue related to the fuel assembly integrity. The USAR claimed that the fuel assembly integrity was maintained during a cask drop based on the Lawrence Livermore Report UCID-21246. A review of the report showed that it evaluated only the cladding, but not any of the other components of the fuel assembly.

The design objectives of the dry storage system are to ensure that the retrieval of fuel assemblies is assured and that fuel rod integrity is not compromised, following the worst case postulated cask drop accident.

Activity Summary: The integrity of fuel assemblies contained within a DSC, following a postulated 75g drop, was analyzed. The drops considered were a horizontal drop, a right-side-up vertical drop, and an upside-down vertical drop. The oblique drop deceleration is bounded by the decelerations of axial and lateral drops. The analyses consisted of an evaluation of the impact of the drop on all of the fuel assembly components, namely the fuel rods, guide tubes, spacer grids, retention grid, and upper and lower end fittings. The objectives of this evaluation were to determine the impact on safety issues, such as confinement, criticality, cladding temperature, and retrievability.

Stress intensities were calculated for the fuel rods, guide tubes, lower end fitting, and retention grid. The calculated values were below the ASME code allowables for their respective materials. Therefore, none of these components would fail following a drop.

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The UEF was analyzed for its limiting scenario of an upside-down vertical drop. The analysis was based on the consideration of elastic-plastic material behavior. It was determined that the UEF ligaments would not fail, and that at least a factor of 2 margin would be available against ductile tearing.

The spacer grids were analyzed for their structural integrity. Both brittle (irradiated) and ductile (un-irradiated) cases were considered in the calculation. It was determined that a major structural damage of the spacer grid would occur for both ductile and brittle material behavior assumptions. The grid failures could allow fuel rods to be relocated from their original grid locations, and create the possibility of one of them getting wedged against the guide sleeve. It was further determined that, with this possibility, an estimated additional pull force of about 220 lbs. would be required for retrieving the fuel assembly from the DSC. The additional force required is within the capacity of the fuel handling machine. Therefore, retrievability of the fuel assembly from the DSC would not be compromised.

The effect of impact of a broken spacer grid fragment, during a horizontal drop, on the fuel rod cladding was investigated. Various cladding vs. fragment orientations and edge conditions were considered. The maximum cladding wall stress was found to be less than the allowable stress.

The failure of spacer grids was determined also to cause a reduction in the fuel rod pitch. The impacts of this change on the criticality and cladding temperature were analyzed. The criticality calculation determined that the effective multiplication factor (k_{eff}) would still be less than 0.95. The cladding temperature evaluation determined that the temperature for the reduced rod pitch would be lower than that for the normal rod pitch because of better conductivity of the new arrangement.

License Amendment Determination: This activity was evaluated against the criteria of 10CFR72.48(a)(2), such as the frequency of occurrence or the consequences of an accident or the malfunction of equipment important to safety. It was concluded that the activity does require a License Amendment for a change to the Technical Specifications, and because it creates the possibility of a malfunction of an SSC important to safety with a different result than that previously evaluated in the ISFSI-USAR.

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ACTIVITY: ES200000755 **50.59 Log No.:** ______ **72.48 Log No.:** SE00155

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Mahmoud Massoud M. Massoud Department: NED Date: 9/6/00
PRINTED NAME AND SIGNATURE

☒ YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: R. H. Beall **PRINTED NAME** **SIGNATURE** Work Group: NFM Date: 9/6/00	Resp. Ind.: **PRINTED NAME** **SIGNATURE** Work Group: Date:	Resp. Ind.: **PRINTED NAME** **SIGNATURE** Work Group: Date:

Approved ☑ Disapproved ☐ Signature I M Sommerville **INDEPENDENT REVIEWER** Date 9-6-00	Approved ☒ Disapproved ☐ Signature **GS-DES, GS-TSES, or PE-PDSU** Date 9/11/00

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 00-077 Date: 9/11/00

Recommend Approval ☑ Recommend Disapproval ☐ Signature ______ Date 9/11/00
POSRC CHAIRMAN

Approved ☑ Disapproved ☐ Signature ______ Date 9/11/00
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ YES ☐ NO
Signature: W T Russell Date: 11/9/00
OSSRC SES Chairman
If yes, OSSRC Meeting No. ______

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ACTIVITY: **ES200000755** **50.59 Log No.:** **72.48 Log No.:** **SE00155**

Proposed Activity: As explained in the attached sheets

Reason for Activity: As explained in the attahced sheets

Function(s) of affected SSC: As explained in the attached sheets

SAR Revision No.: 8

SAR Sections Reviewed: 8.2

Tech Spec Bases Amendment/Rev No.:

Tech Spec Bases Reviewed:

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ Yes ☒ No May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: As explained in the attached sheets

☐ Yes ☒ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: As explained in the attached sheets

☐ Yes ☒ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: As explained in the attached sheets

☐ Yes ☒ No May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident: As explained in the attached sheets

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2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ Yes ☒ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction: As explained in the attached sheets

☐ Yes ☒ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident: As explained in the attached sheets

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ Yes ☒ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
	As explained in the attached sheets

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 4 of 4)

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ACTIVITY: **ES200000755** **50.59 Log No.:** **72.48 Log No.:** **SE00155**

Complete for 72.48:

☐ Yes ☒ No Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

As Explained in the attached sheets.

☐ Yes ☒ No Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

As Explained in the attached sheets.

Summary: (For NRC Report, provide a brief overview)

As explained in the attached sheets

72.48 – SAFETY EVALUATION
ISFSI – SAFETY ANALYSIS UPDATE

PROPOSED ACTIVITY

This safety evaluation is prepared to update Section 8.2.9.2 of SAR of the ISFSI to reflect the modified analysis of the DSC internal pressure during accident condition.

Reason for Activity

Nuclear Engineering Unit procedures require an owner acceptance review of vendor calculations. During such review of TRANSNUCLEAR WEST (TNW) calculation BGE001.0401, "DSC Internal Pressure, Rev. 1" (NEU calculation CA03947) two discrepancies were detected. The first error, dealt with non-conservative pressure, temperature, and volume used to calculate the helium mass for fuel rods. Correct values of 465 psia, 68 F, and 1.99 in^3 now replace the earlier values of 435 psia, 630 F, and 1.23 in^3, respectively. Issue report IR1-043-511 was created to trace the resolution of this change. The second discrepancy deals with the helium temperature used in the backfill process of the Dry Shielded Canister (DSC). The correct temperature of 290 F calculated by NEU through a transient analysis replaces 362 F used in the earlier analysis. The latest revision of the TNW calculation now uses the updated values and calculates a higher helium mass available for DSC pressurization in a design base accident. This results in a higher peak pressure of 64.55 psia, which is still less than the pressure limit of 64.7 psia.

Function of the Affected SSC

The affected SSC is the DSC. The safety functions of DSC are containment, shielding, criticality control, and heat transfer. The primary function of the DSC is to provide containment for the spent nuclear fuel. Hence, the DSC consists of a stainless steel shell, which is hermetically sealed by welding to the two inner top and bottom cover plates. There are also two outer cover plates welded to the shell to ensure containment integrity. The gamma ray shielding at both ends of the DSC is provided by the use of thick lead plugs. This is to minimize exposure from the top during the DSC drying and sealing process. This is also to minimize exposure from the bottom of the DSC while placed in the horizontal storage module (HSM). The shell thickness provides some radial shielding even though shielding in the radial direction is not a safety function of the DSC.

The internal basket assembly of the DSC provides criticality control. The fuel assemblies are maintained in the desired position by a series of guide sleeves and axial support rods during normal and accident conditions. The location and the thickness of the guide sleeves in addition to the distance between the fuel assemblies provide the criticality control. Heat is transferred from the fuel rods to the DSC shell and the cover plates by a combination of thermal radiation, thermal conduction, and internal natural circulation mechanisms. The conduction heat transfer takes place through guide-sleeves, spacer grids, as well as the helium cover gas. The internal circulation takes place by the helium gas between the heat source (fuel rods), and the heat sink (DSC shell and cover plates). The DSC shell in turn is air cooled by natural convection provided by ventilation in the horizontal storage module (HSM). Cooling of the fuel rods is essential in maintaining the integrity of the cladding, which serves as the primary containment barrier of the fission gases and prevents oxidation of uranium. Finally, the DSC integrity must be maintained as it acts as the final barrier for the release of fission gases in to the environment. The design pressure limit of the DSC under accident conditions is 64.7 psig. An accident condition is defined as an event leading to the blockage of all HSM vent paths when ambient temperature is 103 F and 30% of fuel rod gases are released in to the DSC.

Revised Analysis

The DSC internal pressure is a factor of the gas mass in the DSC. The contribution to the gas mass in the DSC consists of three components as follows. First, the backfill helium injected into the DSC subsequent to the fuel assembly loading. Second, the fill helium of fuel rods assumed to enter DSC in a hypothetical failure of all rods. Third, the fuel rod fission gases of which 30% is assumed to enter DSC in a hypothetical failure of all rods. Calculation of the mass of each gas requires three parameters namely, pressure, temperature, and volume as detailed below.

Mass of backfill helium. The ISFSI technical specification 2.2.2 for helium backfill specifies a pressure up to 2.5 psig ± 2.5 psi. A conservative pressure of 5 psig is used in the analysis. As part of this activity the administrative procedure ISFSI-01, "Independent Spent Fuel Storage Installation Procedure" that fills the DSC with helium will be changed to restrict the maximum fill pressure to 2.5 psig. This is to account for 2.5 psi instrument uncertainty. The DSC helium backfill mass is calculated using a DSC volume of 235 ft^3 and a conservatively low temperature of 290 F. The fill temperature is obtained from a transient analysis to ensure the maximum partial pressure is reached due to the injected helium. This results in an additional helium mass of about 9% than originally calculated.

Mass of fuel rod helium. The rod fill helium mass is calculated using revised values for pressure, temperature, and fuel rod gap volume. These are 465 psia for pin pressure, 68 F for gas temperature, and a conservative gap volume of 1.99 in^3 per rod. These changes resulted in a helium mass, which is larger by a factor of about 3.6 compared to the originally calculated helium mass.

Mass of fuel rod fission gases. No discrepancy was found in the calculation of the fission gases. Hence, no change is made and the total moles of the fission gases remain as 194. Using the above mentioned mass of gases, the partial pressure due to the DSC helium backfill is 26.49 psia and the partial pressure due to the fuel rod helium and fission gases is 38.06 psia. Using the design basis accident condition (ambient temperature of 103 F), total DSC pressure reaches 64.55 psia, being below the DSC design pressure of 64.7 psia.

Since the peak pressure in the most conservative circumstance, used in the analysis, is still below the design limit, it can then be concluded that the additional mass of the helium gas does not adversely affect any of the DSC safety functions.

ISFSI USAR Revision No.:......................... 8
ISFSI USAR Section Reviewed:................. 8.2.7. and 8.2.9
ISFS Tech Spec. Bases Amendment/Rev. No.: 1
ISFSI Tech Specification Bases Reviewed: 2.2.2 and 2.4

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR in not increased

Probability of Malfunction
The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. This activity involves making changes to fuel parameters assumed in an ISFSI accident analysis for the fuel, to be stored in the BGE Calvert Cliffs ISFSI facility. Critical functions that must be maintained are containment, shielding, criticality control, and thermal safety functions. The probability of a malfunction is only increased if the performance of any equipment or components required to perform the above functions is degraded. In this case, none of these functions are degraded.

The source strength of the gamma and neutron radiation remain within the original design limits and shielding properties are unchanged, thus shielding functions are not impacted by the changes outlined above. The requirements for heat transfer for the DSC to maintain temperatures below normal operating limits are not affected by these changes. The increase in initial temperature of the helium fill gas does not challenge the DSC's containment function since calculated pressures remain below design limits. The DSC and fuel assembly functions to prevent criticality are not impacted by the changes in fuel parameters since the changes in initial uranium mass and fuel pellet diameter are still bounded by the existing criticality analysis. Thus the probability of any malfunctions is not increased.

Consequence of Malfunction

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. Critical functions that must be maintained are containment, shielding, criticality control, and thermal safety. The probability of a malfunction is increased if the performance of any equipment or components required to perform these functions is degraded. As discussed earlier, none of these functions are degraded. Gamma and neutron source strengths remain within the original design limits and shielding properties are unchanged, thus shielding functions are not impacted by this change to the fuel parameters. Heat transfer functional requirements for the DSC to maintain temperatures below normal operating limits are not affected by these changes. The increase in initial pressure of helium fill gas does not challenge the DSC's containment function since calculated pressures remain below design limits. The DSC and fuel assembly functions to prevent criticality are not impacted by the changes in fuel gap volume. Therefore, there is no increase in the consequences of malfunction of equipment important to safety. There is no release of gases. Hence there is no consequences.

Probability of Accident

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of this activity. Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. Accidents affected by this change in fuel parameters include the drop accident, the accidental pressurization, and the blockage of air inlets and outlets. The changes in fuel parameters do not lead to an increased likelihood of any of these accidents. Assembly weights remain unchanged and within acceptable limits, thus the drop accident probability is not increased. The increased moles of the helium fill gas and the fuel rod gap volume does not increase the probability of accidental pressurization of the DSC. This is because, the calculated peak pressure using conservative inputs and assumptions remains below the design limit. There is no change to the operation of the ISFSI system caused by this activity, therefore there is no change to the probability of any analyzed accident.

Consequences of an Accident

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. Accidents affected by this change in fuel parameters include the drop accident, the accidental pressurization, and the blockage of air inlets and outlets.

Increases in consequences only occur when doses to the public are increased beyond what were previously calculated. Gamma and neutron source strengths remain within the original design limits and shielding properties are unchanged, thus shielding functions are not impacted by this change to the fuel parameters. The increase in initial temperature of the helium fill gas does not challenge the DSC's containment since the calculated peak pressure remains below the design limit. The DSC and fuel assembly functions to prevent criticality are not impacted by the changes in fuel parameters since the change in the fuel rod gap volume has no effect on the criticality analysis. Additionally there are no changes to the operations of the ISFSI created by this activity. Therefore, there is no increase in the consequences of accidents evaluated in the SAR. There is no release of radioactive materials to the environment. Hence, there are no consequences.

2. **The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created**

The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

Possibility of New Malfunction

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. This activity involves a revised analysis of the mass of helium in the DSC, to be stored in the CEG Calvert Cliffs ISFSI facility. Critical functions that must be maintained are containment, shielding, criticality control, and thermal safety functions. All fuel parameters remain within acceptable limits, thus no new malfunctions exist. Since the operation of the ISFSI system is not changed by these changes in fuel parameters, no new malfunctions of equipment or materials are involved.

Possibility of New Accident

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. The operation of the ISFSI system is not changed by the revised analysis discussed above. Criticality is not affected and all other parameters remain within acceptable limits, thus the possibility of a new accident does not exist.

3. **The margin to safety as defined in the basis for any Technical Specification is not reduced**

The margin of safety as defined in the basis for the BGE ISFSI Technical Specification 2.4 is not reduced since the analysis documents that the stress and dose allowables specified in the FSAR and SER are still maintained. Technial Specification 2.4 requirements for HSM dose rates are associated with initially loaded HSMs. Technical specification 2.2.2 is respected in the analysis. Thus there is no Unreviewed Safety Qustion.

Occupational Dose

An increase in occupational dose will not occur as a result of this activity. The operation of the ISFSI system is not changed by the changes determined in the revised analysis. Gamma and neutron source strengths remain within the original design limits and shielding properties are unchanged, thus occupational doses are not affeted.

Environmental Impact

An environmental impact will not occur as the result of this activity. Integrity of the DSC or transfer cask is not affected by this activity. Shielding functions of the DSC and the transfer cask are not affected by this activity. This activity does not affect any area of the plant site previously undisturbed for the ISFSI, and does not cause any reason for revision to the ISFSI Updated Environmental Report. The proposed activity does not affect the environmental conditions associated with the ISFSI.

Summary: (For NRC Annual Report, provide a brief overview)

This activity involves a change in the ISFSI UFSAR to reflect the revised peak pressure calculated for the DSC. This revised peak pressure is larger than before but still below the design limit. No other parameter is affected. The reason for calculating a new peak pressure for the DSC is the fact that the DSC helium mass in increased due to two factors. First, a more conservative temperature is used for helium during the backfill process to maximize the DSC pressure in the most limiting accident scenario. The second factor is the use of a larger fuel rod gap volume, which increases helium mass stored in each fuel rod.

Gamma and neutron source strengths remain within the original design limits and shielding properties are unchanged, thus shielding functions are not impacted by the change in the fuel gap volume. Heat transfer functional requirements for the DSC to maintain temperatures below normal operating limits are not affected by these changes. The increase in initial temperature of the helium fill gas does not challenge the DSC's containment function since calculated pressures remain below design limits. The DSC and fuel assembly functions to prevent criticality are not impacted by the change in fuel gap volume. Assembly weight and overall DSC weights remain within acceptable limits. Thus the changes do not involve a Unreviewed Safety Question.

NUHOMS-24P design, this accident assumes that the fuel rods and the DSC pressure boundary are ruptured due to an event of unspecified origin.

8.2.8.2 Accident Analysis

There are no structural or thermal consequences resulting from the DSC leakage accident described above. The radiological consequences of this accident are presented in Section 8.2.8.3.

8.2.8.3 Accident Dose Consequences

Whole body and maximum organ doses are calculated for a hypothetical individual assumed to be present at the nearest controlled area boundary location (with respect to the ISFSI, approximately 3900') for the duration of the event. A meteorological dispersion parameter (X/Q) of 3.0x10^{-4} sec/m^3 was used in calculating the maximum potential doses at the 3900' controlled area boundary. The resulting calculated doses are 0.1 mrem and 17.8 mrem for the maximum off-site total body and skin doses, respectively. These accident doses are well within the 10 CFR 72.106 limit of 5000 mrem.

8.2.9 ACCIDENTAL PRESSURIZATION OF DRY SHIELDED CANISTER

For more information see Reference 8.16.

This accident addresses the consequences of accidental pressurization of the DSC.

8.2.9.1 Cause of Accident

Internal pressurization of the DSC could result from fuel cladding failure that would release fuel rod fill gas and free fission gas.

8.2.9.2 Accident Analysis

465 psia

The maximum DSC accident pressurization was calculated assuming that the fuel rod fission gas release fraction is 30%, and that the fuel rod fill gas pressure is 435 psia. The resulting internal DSC pressure was calculated at the Calvert Cliffs maximum ambient temperature of 103°F. The limiting accident for DSC pressurization is the HSM blocked vent case discussed in Section 8.2.7. Under these conditions, the gas temperatures in the DSC will rise to 548°F producing a DSC internal pressure of 30.9 psig. The maximum DSC shell local primary membrane stress intensity due to accident pressurization was calculated using 50 psig, the design basis accident pressure discussed in Reference 8.1, and was determined to be 5.1 ksi.

49.85 psig

8.2.9.3 Accident Dose Calculations

Since the maximum DSC accident pressure is within the design basis limits, there are no dose consequences.

8.2.10 FOREST FIRE

For more information see Reference 8.18.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 1 of 4)

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ACTIVITY: ES200000755 **50.59 Log No.:** ____ **72.48 Log No.:** SE00156

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Mahmoud Massoud M. Massoud Department: NED Date: 9/13/00
PRINTED NAME AND SIGNATURE

☒ YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: R. H. Beall **PRINTED NAME**	Resp. Ind.: ____ **PRINTED NAME**	Resp. Ind.: ____ **PRINTED NAME**
SIGNATURE	**SIGNATURE**	**SIGNATURE**
Work Group: NFM	Work Group:	Work Group:
Date: 9/11/00	Date:	Date:

Approved ☒ Disapproved ☐	Approved ☒ Disapproved ☐
Signature ____ **INDEPENDENT REVIEWER**	Signature ____ 9/14/00 **GS-DES, GS-TSES, or PE-PDSU**
Date 9/11/2000	Date 9/14/00

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 00-078 Date: 9/13/00
Recommend Approval ☑ Recommend Disapproval ☐ Signature ____ **POSRC CHAIRMAN** Date 9/13/00

Approved ☑ Disapproved ☐ Signature ____ **PLANT GENERAL MANAGER** Date 9/20/00

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ YES ☒ NO
Signature: ____ Date: 11/9/00
OSSRC SES Chairman
If yes, OSSRC Meeting No. ____

ACTIVITY: ES200000755 **50.59 Log No.:** **72.48 Log No.:** SE00156

Proposed Activity: As explained in the attached sheets

Reason for Activity: As explained in the attahced sheets

Function(s) of affected SSC: As explained in the attached sheets

SAR Revision No.: 8

SAR Sections Reviewed: 8.2

Tech Spec Bases Amendment/Rev No.:

Tech Spec Bases Reviewed:

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ Yes ☒ No May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: As explained in the attached sheets

☐ Yes ☒ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: As explained in the attached sheets

☐ Yes ☒ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: As explained in the attached sheets

☐ Yes ☒ No May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident: As explained in the attached sheets

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 3 of 4)

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ACTIVITY **ES200000755** **50.59 Log No.:** **72.48 Log No.:** **SE00156**

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ Yes ☒ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction: As explained in the attached sheets

☐ Yes ☒ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident: As explained in the attached sheets

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ Yes ☒ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
	As explained in the attached sheets

ACTIVITY: **ES200000755** **50.59 Log No.:** **72.48 Log No.:** **SE00156**

Complete for 72.48:

☐ Yes ☒ No Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose: As explained in the attached sheets

☐ Yes ☒ No Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact: As explained in the attached sheets

Summary: (For NRC Report, provide a brief overview)

As explained in the attached sheets

72.48 – SAFETY EVALUATION
ISFSI – FUEL PARAMETER UPDATE

PROPOSED ACTIVITY

This safety evaluation is prepared to update ISFSI SAR to reflect changes in Tables 1.2-1, 3.3-3, and 3.3-5.

Reason for Activity

This activity is a result of the resolution of IR3-007-603. This issue report documents that Batches A, B, and C used in Units 1 and 2 had different uranium mass, pellet diameter, and clad thickness than specified in various tables of the ISFSI SAR. In response to this issue report, the design basis analysis for ISFSI has been verified as bounding for parameters of Units 1 and 2 fuel Batches A, B, and C. This activity will incorporate the fuel batch A, B, and C parameters into the ISFSI SAR as necessary.

Regarding the uranium mass issue, the IR notes that fuel assemblies contain as much as 399 kg of uranium while only 386 kg of uranium is specified in Table 1.2-1 of the ISFSI SAR. The change to Table 1.2-1 includes replacement of "nominal" to "minimum" for initial uranium content. Regarding the pellet diameter, the IR notes that the actual fuel assembly characteristics for Batches A, B, and C (pellet diameter of 0.3805 inches) are different than that specified in SAR Table 3.3-3 (pellet diameter of 0.3765 inches). Regarding clad thickness, Batches A, B, and C have clad thickness of 0.026 inches. The change to Tables 3.3-3 and 3.3-5 includes a footnote describing that the slight deviation of pellet diameter and clad thickness do not affect the results of design basis analysis.

Function of the Affected SSC

The affected SSC is the DSC. The DSC has containment, shielding, criticality control, and thermal safety functions. The primary function of the DSC is to provide containment for the spent nuclear fuel. Hence, the DSC consists of a stainless steel shell, which is hermetically sealed by welding to the two inner top and bottom cover plates. There are also two outer cover plates welded to the shell to ensure containment integrity. The gamma ray shielding at both ends of the DSC is provided by the use of thick lead plugs. This is to minimize exposure from the top during the DSC drying and sealing process. This is also to minimize exposure from the bottom of the DSC while placed in the horizontal storage module (HSM). The shell thickness provides some radial shielding even though shielding in the radial direction is not a safety function of the DSC.

The internal basket assembly of the DSC provides criticality control. The fuel assemblies are maintained in the desired position by a series of guide sleeves and axial support rods during normal and accident conditions. The location and the thickness of the guide sleeves in addition to the distance between the fuel assemblies provide the criticality control. Heat is transferred from the fuel rods to the DSC shell and the cover plates by a combination of thermal radiation, thermal conduction, and internal natural circulation mechanisms. The conduction heat transfer takes place through guide-sleeves, spacer grids, as well as the helium cover gas. The internal circulation takes place by the helium gas between the heat source (fuel rods), and the heat sink (DSC shell and cover plates). The DSC shell in turn is air cooled by natural convection provided by ventilation in the horizontal housing module (HSM). Cooling of the fuel rods is essential in maintaining the integrity of the cladding, which serves as the primary containment of the fission gases and prevents oxidation of uranium. Finally, the DSC integrity must be maintained as it acts as the final barrier for the release of fission gases in to the environment.

Revised Analysis

Mass of uranium and pellet diameter. As part of the resolution of IR3-007-603, it was noted that different calculations use different values for uranium mass. This is justified, as the goal in all such calculations is to use conservative inputs and assumptions. In a calculation where the goal is to determine the amount of negative reactivity, the use of a low value for uranium mass is justified and conservative. On the other hand, a high value for uranium mass is conservative when the goal was to determine the sources of positive reactivity. In conclusion, the current criticality analysis is bounding for Batches A, B, and C parameters.

Fuel Pellet Diameter. Issue report IR3-007-603 denotes that several fuel assemblies listed in the ISFSI SAR are different from actual fuel assembly characteristics for Batches A, B, and C. Table 3.3-3 and 3.3-5 of ISFSI SAR specifies pellet diameter of 0.3765. The pellet diameter of Batches A, B, and C is 0.3805 inches. Resolution of this issue is based on the fact that, criticality is a function of uranium mass and density. The uranium mass and density in calculation CA03971 (Vectra calculation 113-113.0600) bounds all applicable fuel batches, i.e., for Batches A – G for Units 1 and 2.

Clad thickness. Similar to fuel pellet diameter, there are discrepancies between fuel clad thickness specified in the ISFSI SAR Tables 3.3.-3 and 3.3-5 and clad thickness of fuel Batches A, B, and C. Resolution of this issue is based on the fact that there is no concern regarding reactivity. Clad thickness affects fuel rod gap volume that is considered in SE00155. Clad thickness also affects the allowable pressure stress for a given clad temperature. To ensure that the maximum stress remains below the allowable, and no fuel would fail within the specified ISFSI lifetime and specified fuel burnup, calculation CA03949 (TNW BGE-01.0403, Rev. 1) performs a conservative analysis and establishes a clad temperature limit. The results of new cases analyzed for the thinner cladding associated with Batches A, B and C indicate that the clad temperature limit remains conservative for these batches as well.

ISFSI USAR Revision No.:.........................	8
ISFSI USAR Section Reviewed:.................	Tables 1.2-1, 3.3-3, and 3.3-5
ISFS Tech Spec. Bases Amendment/Rev. No.:	1
ISFSI Tech Specification Bases Reviewed:	-

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR in not increased

Probability of Malfunction

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. This activity involves uranium mass, pellet diameter, and clad thickness. None of these changes would have any affect on the DSC function. Critical functions that must be maintained are containment, shielding, criticality control, and thermal safety functions. The probability of a malfunction is only increased if the performance of any equipment or components required to perform the above functions is degraded. In this case, none of these functions are degraded.

The source strength of the gamma and neutron radiation remain within the original design limits and shielding properties are unchanged, thus shielding functions are not impacted by the changes outlined above. The requirements for heat transfer for the DSC to maintain temperatures below normal operating limits are not affected by these changes. The DSC and fuel assembly functions to prevent criticality are not impacted by the changes in fuel parameters since the changes in initial uranium mass, fuel pellet diameter, and clad thickness are still bounded by the existing criticality analysis. The cladding temperature limit is also not affected. Thus the probability of any malfunctions is not increased.

Consequence of Malfunction

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. Critical functions that must be maintained are containment, shielding, criticality control, and thermal safety. The probability of a malfunction is increased if the performance of any equipment or components required to perform these functions is degraded. As discussed earlier, none of these functions are degraded. Gamma and neutron source strengths remain within the original design limits and shielding properties are unchanged, thus shielding functions are not impacted by this change to the fuel parameters. Heat transfer functional requirements for the DSC to maintain temperatures below normal operating limits are not affected by these changes. The DSC and fuel assembly functions to prevent criticality are not impacted by the changes in fuel parameters since the changes in initial uranium mass, fuel pellet diameter, and clad thickness are still bounded by the existing criticality analysis. The cladding temperature limit is also not affected. Therefore, there is no increase in the consequences of malfunction of equipment important to safety. There is no release of gases. Hence there are no consequences.

Probability of Accident

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of this activity. Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. Accidents affected by this change in fuel parameters include the drop accident, the accidental pressurization, and the blockage of air inlets and outlets. The changes in fuel parameters do not lead to an increased likelihood of any of these accidents. Assembly weights remain unchanged and within acceptable limits, thus the drop accident probability is not increased. The DSC and fuel assembly functions to prevent criticality are not impacted by the changes in fuel parameters since the changes in initial uranium mass, fuel pellet diameter, and clad thickness are still bounded by the existing criticality analysis. The cladding temperature limit is also not affected. There is no change to the operation of the ISFSI system caused by this activity, therefore there is no change to the probability of any analyzed accident.

Consequences of an Accident

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. Accidents affected by this change in fuel parameters include the drop accident, the accidental pressurization, and the blockage of air inlets and outlets.

Increases in consequences occur only when doses to the public are increased beyond what were previously calculated. Gamma and neutron source strengths remain within the original design limits and shielding properties are unchanged, thus shielding functions are not impacted by this change to the fuel parameters. The DSC and fuel assembly functions to prevent criticality are not impacted by the changes in fuel parameters since the changes in initial uranium mass, fuel pellet diameter, and clad thickness are still bounded by the existing criticality analysis.. There are no changes to the operations of the ISFSI created by this activity. Therefore, there is no increase in the consequences of accidents evaluated in the SAR. The cladding temperature limit is also not affected. There is no release of radioactive materials to the environment. Hence, there are no consequences.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created

The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

Possibility of New Malfunction

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. Critical functions that must be maintained are containment, shielding, criticality control, and thermal safety functions. All fuel parameters remain within acceptable limits, thus no new malfunctions exist. Since the operation of the ISFSI system is not changed by these changes in fuel parameters, no new malfunctions of equipment or materials are involved.

Possibility of New Accident

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. The operation of the ISFSI system is not changed by the revised analysis discussed above. Criticality is not affected and all other parameters remain within acceptable limits, thus the possibility of a new accident does not exist.

3. The margin to safety as defined in the basis for any Technical Specification is not reduced

The margin of safety as defined in the basis for the BGE ISFSI Technical Specification 2.4 is not reduced since the analysis documents that the stress and dose allowable specified in the FSAR and SER are still maintained. Thus there is no Unreviewed Safety Question. Regarding the mass of uranium, pellet diameter, and clad thickness, the margin to safety is not reduced. This is true because the analyses of record remain bounding.

Occupational Dose

An increase in occupational dose will not occur as a result of this activity. The operation of the ISFSI system is not changed by the changes determined in the revised analysis. Gamma and neutron source strengths remain within the original design limits and shielding properties are unchanged, thus occupational doses are not affeted.

Environmental Impact

An environmental impact will not occur as the result of this activity. Integrity of the DSC or transfer cask is not affected by this activity. Shielding functions of the DSC and the transfer cask are not affected by this activity. This activity does not affect any area of the plant site previously undisturbed for the ISFSI, and does not cause any reason for revision to the ISFSI Updated Environmental Report. The proposed activity does not affect the environmental conditions associated with the ISFSI.

Summary: (For NRC Annual Report, provide a brief overview)

This activity is a result of the resolution of IR3-007-603. This issue report documents that batches A, B, and C used in units 1 and 2 had different uranium mass, pellet diameter, and clad thickness than specified in various tables of the ISFSI SAR. In response to this issue report, the design basis analysis for ISFSI have been verified as bounding for parameters of units 1 and 2 fuel batches A, B, and C. This activity will incorporate the fuel batch A, B, and C parameters into the ISFSI SAR as necessary.

Regarding the uranium mass issue, the IR notes that fuel assemblies contain as much as 399 kg of uranium while only 386 kg of uranium is specified in Table 1.2-1 of the ISFSI SAR. The change to Table 1.2-1 includes replacement of "nominal" to "minimum" for initial uranium content. Regarding the pellet diameter, the IR notes that the actual fuel assembly characteristics for Batches A, B, and C (pellet diameter of 0.3805 inches) are different than that specified in SAR Table 3.3-3 (pellet diameter of 0.3765 inches). The change to Tables 3.3-3 and 3.3-5 includes a footnote describing that the slight deviation of pellet diameter and clad thickness do not affect the results of design basis analysis.

Gamma and neutron source strengths remain within the original design limits and shielding properties are unchanged, thus shielding functions are not impacted by the change in the fuel parameters. Heat transfer functional requirements for the DSC to maintain temperatures below normal operating limits are not affected by these changes. The DSC and fuel assembly functions to prevent criticality are not impacted by the changed fuel parameters. The cladding temperature limit is also not affected. Assembly weight and overall DSC weights remain within acceptable limits. Thus the changes do not involve a Unreviewed Safety Question.

TABLE 1.2-1

DESIGN PARAMETERS FOR THE CALVERT CLIFFS ISFSI

GENERAL DESIGN REQUIREMENTS

Capacity (Fuel Assemblies/Canister)		24 Pressurized Water Reactor Assemblies
Reference Fuel Assembly Parameters:		
Burnup: Max. Assembly Average		47,000 MWD/MTU
Initial Enrichment (Maximum)		4.5 w/o U^{235}
Initial Uranium Content (~~Nominal~~)	Minimum	386 kg/Assembly
Decay Heat Power (Maximum)		0.66 kW/Assembly
Cooling Time		As Required for Decay Heat Limit
Fuel Rod Array		Combustion Engineering 14x14
Assembly Weight (Maximum)		1,300 lbs
Maximum Assembly Envelope		8.25 in^2
Effective Multiplication Factor:		
Normal		$K_{eff} < 0.95$
Off-Normal		$K_{eff} < 0.98$
Internal DSC Atmosphere		Helium
Ambient Temperature Range		-3°F to 103°F
Solar Heat Load:	Maximum	127 Btu/hr-ft^2
	Average	82 Btu/hr-ft^2
Nominal Dose at HSM Surface During Storage (Away from Openings)		20 mrem/hr
Maximum Dose at HSM Door and Penetrations		100 mrem/hr
Peak Long-Term Clad Temperature (70°F Ambient)		612°F
Credit for Burnup Criticality Analysis		Based on 1.8% equivalent initial enrichment
Maximum Assembly Length (Includes Radiation Growth)		less than 158.0"
Active Fuel Length		136.7"

TABLE 3.3-5

DESIGN PARAMETERS FOR CRITICALITY ANALYSIS OF THE DSC

PARAMETERS	DESIGN VALUE
FUEL ASSEMBLIES	
Number/Type	24/CE design 14x14
Rod Array	14x14
Number of Fuel Rods	176
Number of Control Rod Guide Tubes	5
Number of Instrument Tubes	1 (1 of the 5 Guide Tubes)
Rod Pitch (inches)	0.580
Burnup Credit	0 - 45 GWD/MTU
FISSILE CONTENT (% initial U equivalent)	
U^{235}	1.8 - 4.5
FUEL PELLETS	
Density	95% Theoretical
Diameter (inches)	0.3765 *
FUEL ROD CLADDING	
Material	Zircaloy - 4
Thickness (inches)	0.028*
Outside Diameter (inches)	0.440
CONTROL ROD GUIDE TUBES	
Material	Zircaloy - 4
Thickness (inches)	0.040
Outside Diameter (inches)	1.115
INSTRUMENT TUBE	
Material	Zircaloy - 4
Thickness (inches)	0.040
Outside Diameter (inches)	1.115
DSC GUIDE SLEEVES	
Material	Stainless Steel
Thickness (inches)	0.1050
DSC FILL MATERIAL	
Material	Pure Water
Density (g/cm^3)	0.0 - 1.0
DSC SHELL	
Material	Stainless Steel
Thickness (inches)	0.625
Outside Diameter (inches)	67.25
CASK	
Material	Stainless Steel/Lead
Thickness (inches)	6.25[a]
Outside Diameter (inches)	80.5[a]

For more information see Reference 3.14.

[a] Exclusive of the Cask Neutron Shield

TABLE 3.3-3

CE 14x14 FUEL PARAMETERS

FUEL ASSEMBLY PARAMETER	INCHES
Fuel Clad OD	0.44
Fuel Pellet OD	0.3765 *
Clad Thickness	0.028 *
Fuel Rod Pitch	0.58
Guide Tube OD	1.115
Guide Tube Thickness	0.04
Active Fuel Height	136.7
Fuel Rods/Assembly	176
No. of Guide Tubes	5

Reference: J. W. Roddy, "Physical and Decay Characteristics of Commercial LWR Spent Fuel," ORNL/TM-9591/V1&R1, January 1988.

For more information see Reference 3.14.

* The fuel pellet OD and clad thickness varied slightly for fuel batches A, B, and C in Units 1 and 2. These variances do not affect the results of design basis analysis.

SAFETY EVALUATION FORM

Page 1 of 11

ACTIVITY: ES200000900	**50.59 Log No.: N/A**	**72.48 Log No.: SE00157**

ISFSI – Evaluation of Transfer Cask Integrity

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Mohammed Kaiseruddin
PRINTED NAME AND SIGNATURE
Department: Sargent & Lundy
Date: 10/18/00

☒ YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye **PRINTED NAME**	Resp. Ind.: C. J. Dobry **PRINTED NAME**	Resp. Ind.: R. H. Beall **PRINTED NAME**
SIGNATURE	**SIGNATURE**	**SIGNATURE**
Work Group: **Licensing**	Work Group: **PES**	Work Group: **NFM**
Date: 10/18/00	Date: 10/19/00	Date: 10/18/00

Approved ☑ Disapproved ☐
Signature J.E. REMENIUK
INDEPENDENT REVIEWER
Date 10.19.00

Approved ☑ Disapproved ☐
Signature
W.E. KEMPER S-DES, GS-TSES, or PE-PDSU
Date 10.20.00

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 00-092 Date: 10/25/00

Recommend Approval ☑ Recommend Disapproval ☐ Signature
POSRC CHAIRMAN
Date 10/27/0

Approved ☑ Disapproved ☐ Signature
PLANT GENERAL MANAGER
Date 10/26/0

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ YES ☒ NO
Signature: W.R. Russell Date: 01/12/01
OSSRC SES Chairman
If yes, OSSRC Meeting No.

ACTIVITY: ES200000900	50.59 Log No.: N/A	72.48 Log No.: SE00157

ISFSI – Evaluation of Transfer Cask Integrity

Proposed Activity:

The proposed activity consists of making changes to the ISFSI USAR [Refs. 1 and 2]. The changes are based upon the re-evaluation of Transfer Cask (TC) structural integrity, which was undertaken because of some concerns in the existing analysis that were identified during an internal review. The new analysis is documented in Baltimore Gas and Electric (BGE) calculation CA04141 [Ref. 7].

The changes to the USAR consist of the following:

1. Section 8.1.1.9 (C) is being replaced by the following text; " Transfer Cask thermal loads are calculated using an axisymmetric cask model. A fuel assembly decay heat power of 15.8 kW was applied as a uniform heat flux to the transfer cask inner surfaces. Convection coefficients, applied as surface loads to the cask outer surfaces, are based on simplified equations for heat loss from various surfaces to air. They are: 0.0066 BTU/hr-in^2-^{o}F, for the cylindrical shell, and 0.0051 BTU/hr-in^2-^{o}F through the cask and plates. Two bounding ambient temperature cases are considered, consisting of –3 ^{o}F and 103 ^{o}F, representing the site-specific historical extremes. A bounding solar heat flux of 62 BTU/hr-ft^2 is also applied for the hot ambient case."
2. Section 8.2.5.2 is being revised to provide a better description of transfer cask drop analysis as, " using the ANSYS 3-D transfer cask one-half model. For the vertical, horizontal, and corner drop orientations, the contacting surface was assumed to be rigid. A static equivalent load of 75g was applied. The internal loading of the DSC was represented as pressure loadings applied to the transfer cask inner surfaces."
3. Table 3.6-1 is being revised to provide site-specific design loads for the TC upper trunnion.
4. Table 8.1-1 is being revised.
5. Tables 8.2-14 through 8.2-16 are being updated with the new calculated stress values and the allowables.
6. Section 8.4 is being revised to add Reference 8.33.

Background

The Independent Spent Fuel Storage Installation (ISFSI) at the Calvert Cliffs Nuclear Power Plant (CCNPP) stores spent fuel assemblies within Dry Storage Canisters (DSCs). Twenty-four spent fuel assemblies are loaded into each DSC. Each DSC contains an outer leak-tight shell and an internal basket assembly. The outer shell provides the structural strength, shielding, and a leak-tight chamber for containing helium. The internal basket assembly includes twenty-four stainless steel guide sleeves (one for each spent fuel assembly), nine perforated carbon or stainless steel spacer discs, and four stainless steel support rods. The nine spacer discs are spaced out along the length of the DSC at locations that approximately coincide with the spent fuel assembly's eight spacer grids and the single lower retention grid. The spacer discs are not structurally attached to the DSC shell walls or inner cover plates. The guide sleeves traverse the length of the DSC cavity through openings in the spacer discs. The support rods are used to maintain the spacer disc locations. The support rods traverse the length of the DSC cavity through the spacer discs, and are structurally welded to the spacer discs. The DSC is loaded into a Transfer Cask (TC) for transporting it to the storage facility.

CCNPP utilizes the Combustion Engineering 14X14 fuel assemblies. Each of the fuel assemblies is approximately 157 inches in length and approximately 8X8 inches square in cross section.

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ISFSI – Evaluation of Transfer Cask Integrity

New Analysis

A new analysis was prepared to re-evaluate the TC structural integrity.

Reference 11 provides the maximum dry weights of the regular and with the value-added pellets (VAP) spent fuel assemblies. CCNPP will be using the VAP fuel assemblies in the future. The new fuel assemblies are approximately 30 lbs. heavier than the old ones. Reference 11 lists the maximum mass of a spent fuel assembly as 1360 lbs., and the weight of a control element assembly (CEA) as 80 lbs. Since the fuel assemblies can be moved and stored with the CEAs inserted in them, the maximum fuel assembly weight used in the analysis was 1450 lbs., which allows for some margin for uncertainties. It is noted that CCNPP is not permitted by its Tech Specs to load into ISFSI any fuel assemblies that are heavier than 1300 lbs. The use of the higher mass in the analysis at this time is conservative, and will be applicable later when CCNPP seeks permission to load the higher mass fuel assemblies into the ISFSI. Using the nominal dimensions of the Dry Shielded Canister (DSC) and Transfer Cask (TC) components, and standard material densities, the final weights of the components were calculated and documented in References 6 and 16. The weight of a DSC sealed and fully loaded with the VAP fuel assemblies was calculated to be 69,400 lbs., which is more than the nominal value used earlier of about 65,000 lbs., but less than the enveloping value used of 80,000 lbs.

The TC integrity analysis is documented in References 7 and 16. The analysis utilized the ANSYS 3-D transfer cask one-half model. For the on-site transfer in normal operating conditions, the design parameters considered for the upper trunnion analysis were +/- 1g vertically, +/- 1g axially, +/- 1g laterally, and (+/- 1/2g vertically +/- 1/2g axially +/- 1/2g laterally). For the vertical, horizontal, and corner drop orientations, the contacting surface was assumed to be rigid. A static equivalent load of 75g was applied. The internal loading of the DSC was represented as pressure loadings applied to the Transfer Cask inner surfaces. Results of the calculations showed that, in all cases, the calculated stresses for the various parts of the TC remained below the code allowables. It is noted here that in Tables 8.2-14 through 8.2-16, the allowable stress values for some of the sub-components are being reduced. This represents only a correction. The values provided earlier were good for the carbon steel. The new values are good for the stainless steel, of which the subject sub-components are made.

Reason for Activity:

This activity incorporates into the USAR the resolution to Issue Reports IR3-005-169 and IR3-005-172 [Refs. 3 and 4]. These issue reports identified inconsistencies in the structural analysis of the TC. A new analysis was prepared [Ref. 7] to replace the one with inconsistencies.

The design objectives of the dry storage system are to ensure that in normal operating conditions the spent fuel can be stored safely without adverse consequences to the operators, public, and the environment. The design objectives in abnormal and accident conditions, such as following the worst case postulated cask drop accident, are that the fuel rod integrity is not compromised and that the retrieval of fuel is still assured.

Technical Specifications Section 2.3, "Transfer Cask", states that the transfer cask lifting height with a non-single-failure-proof lifting device shall not exceed 80 inches. The Technical Specification also states that for drops greater than 15 inches the DSC will be returned to the spent fuel pool and be visually inspected. Therefore, retrievability of fuel from the DSC is demonstrated following a TC drop accident.

Function(s) of affected SSCs:

The affected SSCs are the fuel assemblies, DSCs, and TCs.

Fuel Assembly

The fuel assembly consists of 176 fuel and poison rods, 5 guide tubes, 5 guide tube sleeves, 8 fuel rod spacer grids, upper and lower end fittings, lower retention grid, and a hold-down device. The guide tubes, spacer grids, and end fittings form the structural frame of the assembly. The fuel rod spacer grids maintain the fuel rod pitch over the length of the assembly. The grid provides positive side restraint to the fuel rod but only frictional restraint axially. The spacer grids are the widest part on a fuel assembly. The four outer guide tubes are mechanically attached to the end fittings and the spacer grids are welded to all five guide tubes. The upper end fitting attaches to the guide tubes to serve as an aligning and lifting device for each fuel assembly. The fuel rods are held together in the assembly at a pitch of 0.58 inch, which helps to control the criticality. The rods consist of enriched uranium fuel pellets stacked within a Zircalloy cladding tube. The cladding is the first barrier that prevents the fission products from escaping to the outside.

Dry Storage Canister

The DSC is classified as important-to-safety per 10 CFR 72. It consists of the outer canister and the internal basket assembly. The sub-components of the internal basket assembly include the Spacer Discs, Support Rods, and Guide Sleeves. The internal basket assembly components are not attached structurally to the outer canister.

The DSC provides containment, shielding, criticality control, configuration control related to fuel retrievability, structural support, and thermal safety functions during loading operations, transfer operations, and storage. It is designed to remain intact under all accident conditions identified in the ISFSI USAR with no loss of function. Specific design functions of the DSC include the following:

1. Confinement - The DSC design provides mechanical confinement of the stored fuel assemblies to prevent the dispersion of particulate or gaseous radionuclides from the fuel. The primary function of the DSC is to provide confinement of the spent nuclear fuel. This is achieved by the stainless steel shell and two inner cover plates (top and bottom ends) which are welded to the shell assembly. There are also outer cover plates (top and bottom) to further assure containment integrity. The DSC confinement boundary is designed also to retain helium cover gas around the fuel in order to prevent corrosion of the fuel cladding and formation of expansive oxides in the fuel during storage.
2. Criticality Control - The DSC design provides for sub-criticality during the wet loading, DSC drying, and interim storage operations. This is accomplished by a combination of mechanical separation of the fuel assemblies by the internal basket assembly and neutron absorption in the stainless steel guide sleeve.
3. Fuel Support and Configuration Control - The DSC internal basket assembly provides support for the spent fuel assemblies during normal operations. The DSC also provides configuration control related to post accident recovery of spent nuclear fuel. The DSC is designed so that the worst-case postulated accidents, including a cask drop, will not result in deformation of the Internal Basket Assembly or the DSC shell to such a degree that retrieval of fuel is not assured.
4. Shielding - The DSC materials provide gamma radiation shielding. The DSC provides gamma shielding at its ends by the use of lead shield plugs. These provide ALARA dose rates at the top of the canister during drying and sealing operations and at the bottom for minimizing dose rates during DSC loading into the Horizontal Storage Module (HSM) and at the HSM door during storage.
5. Thermal - Decay heat is removed by thermal radiation and conduction from the DSC to the TC, and by thermal radiation and conduction and convection from the DSC to the HSM. The DSC maintains

the helium cover gas, which is required for corrosion control. This cover gas improves the thermal performance of the DSC.

The functions of the internal basket assembly components are as follows:

6. Guide Sleeves – The guide sleeves establish storage compartments for 24 spent fuel assemblies within the DSC. The tops of the guide sleeves are flared to assist fuel-handling operators in guiding the spent fuel assemblies into the sleeves.
7. Spacer Discs – The spacer discs work together with the guide sleeves to maintain geometric separation of the fuel assemblies. The spacer discs support the weight of the guide sleeves, support rods and the spent nuclear fuel when the DSC is in a horizontal orientation.
8. Support Rods – The support rods maintain the spacer disk locations along the length of the DSC. They carry the weight of the guide sleeves, and the spacer discs when the DSC is in a vertical orientation.

Transfer Cask

The TC is important to safety because it protects the DSC during handling. The TC is also considered safety-related since dropping a loaded TC, which weighs about 100 tons, has the potential for creating unanalyzed accident situations in the power plant. The TC is a cylindrical vessel with a bottom end closure assembly and a bolted top cover plate. Its cylindrical walls are formed from three concentric steel shells, the middle of which is the structural shell. Lead is poured between the inner and the middle shells to provide gamma shielding, and a solid neutron shielding material is poured between the middle and the outer shells. Two upper trunnions are located near the top of the cask, and are used for lifting. Two lower trunnions serve as the axis of rotation of the TC, and provide support.

The solid neutron shielding material used in the cask outer annulus, and top and bottom covers produces water vapor and a small quantity of non-condensable gases when heated to above 212 °F. The off gassing produces an internal pressure, which increases with temperature.

SAFETY EVALUATION FORM

ACTIVITY: ES200000900	50.59 Log No.: N/A	72.48 Log No.: SE00157

ISFSI – Evaluation of Transfer Cask Integrity

ISFSI USAR Revision No.: 9

ISFSI USAR Sections Reviewed:

The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key Sections reviewed are listed as follows:

1.2.1	General Description
4.2.1	Structural Specifications
8.1.1	Normal Operations Structural Analysis
8.2	Accidents
8.2.5	Cask Drop
8.4	References
Table 1.2-1	Design Parameters for the Calvert Cliffs ISFSI
Table 3.1-1	Principal Design Parameters for Fuel to Be Stored
Table 3.3-3	CE 14X14 Fuel Parameters
Table 3.6-1	Summary of Design Criteria for Normal Operating Conditions
Table 3.6-3	Summary of Design Criteria for Accident Conditions
Table 8.1-1	Estimated Component Weights
Table 8.2-1	NUHOMS-24P Accident Loading Identification
Table 8.2-14	NUHOMS-24P Transfer Cask Enveloping Load Combination Results for Normal and Off-Normal Loads (ASME Service Levels A and B)
Table 8.2-15	NUHOMS-24P Transfer Cask Enveloping Load Combination Results for Accident Loads (ASME Service Level C)
Table 8.2-16	NUHOMS-24P Transfer Cask Enveloping Load Combination Results for Accident Loads (ASME Service Level D)

Tech Spec Bases Amendment/Rev No.: 2

Calvert Cliffs Independent Spent Fuel Storage Installation Technical Specifications, Appendix A to Materials License No. SNM-2505, Amendment 2, June 30, 2000

Tech Spec Bases Reviewed:

2.3	Transfer Cask (TC)
3/4.1	Fuel to be Stored at ISFSI
5.0	Design Features

ACTIVITY: ES200000900	**50.59 Log No.: N/A**	**72.48 Log No.: SE00157**

ISFSI – Evaluation of Transfer Cask Integrity

Complete for 50.59 and 72.48:

1. **The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.**

☐ YES ☒ NO **May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Probability of Malfunction:

The proposed activity consists of re-evaluating the structural integrity of TC, and incorporating the results into the USAR. The TC is important to safety because it protects the DSC during handling. The TC is also considered safety-related since dropping a loaded TC, which weighs about 100 tons, has the potential for creating unanalyzed accident situations in the power plant. TC is also designed to provide gamma and neutron shielding in order to keep personnel doses as low as reasonably achievable (ALARA) during the DSC transfer to Horizontal Storage Module (HSM).

The ISFSI equipment, whose functions may be impacted by the lack of structural integrity of the TC, are the fuel assemblies, DSC and TC. The fuel assemblies and DSCs are designed to withstand the accident and abnormal loads, which are much larger than the normal dead weight and handling loads.

The probability of malfunction of equipment important to safety is not impacted by the re-analysis of TC structural integrity.

☐ YES ☒ NO **May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Consequences of Malfunction:

The malfunctions to be considered are of the ISFSI important-to-safety components listed above.

The consequences of failure of the fuel assemblies, DSC and TC are all related to the release of radioactivity into the atmosphere or the dose to operators or the public. The fuel characteristics, such as the decay heat level and radiation dose criteria applicable to ISFSI components, are not impacted by the re-analysis of TC structural integrity.

☐ YES ☒ NO **May the probability of occurrence of an accident previously evaluated in the SAR be increased?**

Probability of Accident:

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. They consist of loss of shielding, external missiles, earthquake, flood, cask drop, lightning, blockage of air inlets and outlets, DSC leakage, DSC overpressurization, and forest fire. Of these accidents, only the cask drop accident and earthquake incident are impacted by this activity. The earthquake scenario is bounded by the cask drop accident, as the acceleration postulated in a design basis earthquake are 0.25g horizontally and 0.17 vertically, which are much smaller than the acceleration in the drop accident of 75g. However, the probability of occurrence of cask drop, or any other accident, is not impacted by the TC integrity analysis.

ACTIVITY: ES200000900	50.59 Log No.: N/A	72.48 Log No.: SE00157

ISFSI – Evaluation of Transfer Cask Integrity

There is no change to the design or operation of the NUHOMS system caused by this activity. This activity does not modify the external configuration of the DSC envelope. The interface between the DSC and the HSM during ISFSI operations and interim storage of the DSC remains unaffected. Therefore, the probability of occurrence of an accident involving loss of HSM air outlet shielding, or blockage of HSM air inlets and outlets will not increase.

Pressurization of the DSC due to fuel cladding failure is an accident scenario identified in USAR Section 8.2.9. The limiting DSC pressurization accident event is a rupture of fuel cladding together with blockage of the HSM vents. The impact of the cask drop on the cladding was evaluated previously, and it was determined that the fuel cladding would not rupture and fuel rod integrity would be maintained.

DSC leakage is an accident scenario described in USAR Section 8.2.8. The USAR indicates that there are no credible events that would initiate this type of accident. As stated in the preceding paragraphs, the probability of an accident that would lead to cladding failure is not increased by this activity. This activity does not affect the design of the DSC pressure boundary and therefore does not increase the probability of DSC leakage.

☐ YES ☒ NO **May the consequences of an accident previously evaluated in the SAR be increased?**

Consequences of Accident:

The proposed activity, namely the evaluation of TC integrity, is related to the cask drop accident and earthquake incident, as stated above.

The ISFSI important-to-safety components that could be impacted by the TC integrity are the fuel assemblies, DSC, and TC. The evaluation of the fuel assembly integrity is not included in the scope of this change; a separate safety evaluation for it is being prepared [Ref. 8]. The evaluation of DSC structural integrity was addressed separately in earlier safety evaluations [Refs. 9 and 10]. The TC was evaluated for a design basis vertical and horizontal drops of 75g. The calculated stresses in TC sub-components were all found to be below the code allowables. Therefore, the TC would not fail, would continue to perform its design functions, and the consequences of the accident would not be increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ YES ☒ NO **May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?**

Possibility of New Malfunction:

The proposed activity evaluates the structural integrity of TC. It does not impact any of the thermal or environmental parameters that would affect other equipment important-to-safety. Therefore, there is no possibility created of a new malfunction in any of the important-to-safety ISFSI components.

☐ YES ☒ NO **May the possibility of an accident of a different type than any previously evaluated in the SAR be created?**

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ISFSI – Evaluation of Transfer Cask Integrity

Possibility of New Accident:

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the USAR, and have been discussed previously. Evaluation of the structural integrity of the TC showed that the important-to-safety components of ISFSI would maintain their safety functions. The impacts of the drop accident on components, such as the fuel assemblies, and DSC were evaluated previously. The results showed that none of the components would fail to perform their safety functions. The impacts on the TC were evaluated with similar results. Since there is no change to the design or operation of the NUHOMS system caused by this activity, the possibility of an accident of a different type than any previously evaluated in the SAR would not be created.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ **YES** ☒ **NO** **Will the margin of safety as defined in the basis for any Technical Specification be reduced?**

Tech Spec Bases: 2.3

Discussion of why the margin of safety is not reduced:

The margin of safety is defined as range of values between the acceptance limit reviewed and approved by the NRC as part of the licensing basis and the failure point [Ref. 14]. USAR Section 3.2.5 defines the acceptance criteria for ISFSI components, none of which would be exceeded. It is noted here that in Tables 8.2-14 through 8.2-16, the allowable stress values for some of the sub-components are being reduced. This represents only a correction. The values provided earlier were good for the carbon steel. The new values are good for the stainless steel, of which the subject sub-components are made.

Therefore, the margin of safety would not be reduced.

Complete for 72.48:

☐ **YES** ☒ **NO** **Will the proposed activity involve a significant increase in occupational dose?**

A significant increase in occupational dose:

The radiation protection design and operation of the NUHOMS-24P dry cask storage system would not be changed by this proposed activity. The analysis of the TC under the design basis accidental drop conditions showed that it would not impact the radioactivity confinement boundary or the criticality control. Retrievability of fuel following a drop would still be assured. Occupational dose associated with post DSC accident recovery is not addressed in the Calvert Cliffs ISFSI USAR, however, the occupational dose for fuel retrieval, following the unlikely cask drop accident, is expected to be minimized through the use of special procedures and precautions. Because none of these attributes would be changed, the occupational doses summarized in USAR Table 7.4-1 would not be affected by this activity.

SAFETY EVALUATION FORM

ACTIVITY: ES200000900	50.59 Log No.: N/A	72.48 Log No.: SE00157

ISFSI – Evaluation of Transfer Cask Integrity

☐ YES ☒ NO **Will the proposed activity involve a significant unreviewed environmental impact?**

A significant unreviewed environmental impact:

The NUHOMS-24P dry cask storage system confinement and radiological shielding functions would not be reduced by this activity. The fuel assemblies, DSC and TC were determined to maintain their safety functions under the most severe postulated drop accident conditions.

This activity would not affect any area of the plant site previously undisturbed for the ISFSI, and would not cause any reason for revision to the ISFSI Updated Environmental Report. This activity would not affect the environmental conditions associated with the ISFSI. Therefore, this activity would not involve an unreviewed environmental impact.

References:

1. Calvert Cliffs Independent Spent Fuel Storage Installation USAR, Rev. 9
2. CCNPP ISFSI USAR Change Request, UCR00193
3. BGE Issue Report IR3-005-169, 02/10/1998
4. BGE Issue Report IR3-005-172, 05/11/1998
5. BGE Calculation 8067-16-10-6, Rev. 1, Fuel Assembly Weight
6. BGE Calculation CA03988, Rev. 0001, Final Weight Calculation of NUHOMS-24P DSC / TC System
7. BGE Calculation CA04141, Rev. 0001, ISFSI Transfer Cask Structural Evaluation
8. BGE 10CFR72.48 Safety Evaluation, SE00154 (Under Preparation)
9. BGE 10CFR72.48 Safety Evaluation, SE00148
10. BGE 10CFR72.48 Safety Evaluation, SE00146
11. BGE Memorandum NEU 99-164, from T. A. Schearer to G. V. Patel, et al., 7/7/1999
12. Technical Specifications for Calvert Cliffs ISFSI, Amendment 2
13. Calvert Cliffs Updated Final Safety Analysis Report (UFSAR), Rev. 26
14. NEI 96-07, Rev. 0, Guidelines for 10 CFR 50.59 Safety Evaluations, 09/97
15. Calvert Cliffs ISFSI Updated Environmental Report, Rev. 1
16. Hopper & Associates Letter HABGE-09/00-0893, Fuel Assembly Weight Issue, 9/25/00

ACTIVITY: ES200000900	50.59 Log No.: N/A	72.48 Log No.: SE00157

ISFSI – Evaluation of Transfer Cask Integrity

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: The proposed activity consists of making changes to the ISFSI USAR. The changes are based upon the re-evaluation of Transfer Cask (TC) structural integrity, which was undertaken because of some concerns in the existing analysis that were identified during an internal review. The new analysis is documented in Baltimore Gas and Electric (BGE) calculation CA04141.

The USAR is being revised to describe the TC re-analysis methodology and modeling, and update the calculated results presented in Chapter 8 Tables.

Reason for Activity: Inconsistencies were identified in the structural analysis of the TC during an internal review. A new analysis was prepared to resolve the inconsistencies and replace the old analysis.

The design objectives of the dry storage system are to ensure that in normal operating conditions the spent fuel can be stored safely without adverse consequences to the operators, public, and the environment. The design objectives in abnormal and accident conditions, such as following the worst case postulated cask drop accident, are that the fuel rod integrity is not compromised and that the retrieval of fuel is still assured.

Activity Summary: New analysis was prepared to re-evaluate the structural integrity of the TC.

Using a conservative fuel assembly mass of 1450 lbs., the nominal dimensions of the Dry Shielded Canister (DSC) and Transfer Cask (TC) components, and standard material densities, the final weights of the components were calculated. The weight of a DSC sealed and fully loaded with fuel assemblies was determined to be less than the enveloping value used earlier of 80,000 lbs.

The ISFSI important-to-safety components that could be impacted by the TC integrity are the fuel assemblies, DSC, and TC. The evaluation of the fuel assembly integrity is the subject of a separate safety evaluation. The evaluation of DSC structural integrity was addressed separately in earlier safety evaluations. The TC was evaluated for a design basis vertical and horizontal drops of 75g. The calculated stresses in TC sub-components were all found to be below the code allowables.

USQ Determination: This activity was evaluated against the criteria of 10CFR72.48(a)(2), such as the probability of occurrence or the consequences of an accident or the malfunction of equipment important to safety, and it was concluded that it does not involve an unreviewed safety question (USQ).

SAFETY EVALUATION FORM

Page 1 of 20

ACTIVITY: ES199800312	**50.59 Log No.: N/A**	**72.48 Log No.: SE00158**

ISFSI – Disposition of ITR Issues and Other USAR Changes

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Mohammed Kaiseruddin
PRINTED NAME AND SIGNATURE
Department: Sargent & Lundy
Date: 01/02/01

☒ YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye **PRINTED NAME**	Resp. Ind.: C. J. Dobry **PRINTED NAME**	Resp. Ind.: R. H. Beall **PRINTED NAME**
SIGNATURE	**SIGNATURE**	**SIGNATURE**
Work Group: **Licensing**	Work Group: **PES**	Work Group: **NFM**
Date: 1/8/01	Date: 1/19/01	Date: 1/18/01

Approved ☒ Disapproved ☐	Approved ☑ Disapproved ☐
Signature J.E. REMENIUK **INDEPENDENT REVIEWER**	Signature **P.D. PATEL** Pravin Patel **GS-DES, GS-TSES, or PE-PDSU**
Date 1·19·01	Date 1/23/01

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 01-006 Date: 01/29/01

Recommend Approval ☑ Recommend Disapproval ☐ Signature **POSRC CHAIRMAN** Date 1/29/01

Approved ☑ Disapproved ☐ Signature **PLANT GENERAL MANAGER** Date 1/29/01

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ YES ☒ NO
Signature: Date: 7/12/01
OSSRC SES Chairman
If yes, OSSRC Meeting No.

Proposed Activity:

The proposed activity consists of making changes to the ISFSI USAR [Refs. 1 and 2]. The changes are being made to:

1. resolve discrepancies in the USAR identified during an Independent Technical Review (ITR) [Ref. 3],
2. correct the DSC design temperature value to 460 °F, and
3. delete the specific calculated values of individual component weights, stresses and temperatures.

The proposed activity does not involve any hardware changes.

For the ease of discussion and evaluation, the proposed changes to the USAR can be categorized into six categories, as described in the table below.

Change Category	Subject	Description	Supporting Analyses
1	Load Combinations	Load combinations used in the structural analyses of ISFSI components have been re-defined using site specific information.	Refs. 4 and 5
2	Transfer Cask (TC) Structural Analysis	The transfer cask structural integrity was re-analyzed. This re-analysis was evaluated for safety previously as 72.48 safety evaluation number SE00157. Some of the related USAR changes are incorporated through the current safety evaluation.	Refs. 7 and 10
3	Thermal/ Pressure Parameters	Discrepancies identified in the thermal/ pressure parameters, through the independent technical review (ITR), are resolved.	Refs. 4 and 5
4	Dry Shielded Canister (DSC) Structural Analysis	The DSC structural integrity was re-analyzed. This re-analysis was evaluated for safety previously as 72.48 safety evaluation numbers SE00133, SE00146, and SE00148. Some of the related USAR changes are incorporated through the current safety evaluation.	Refs. 6, 8, and 9
5	Deletion of Calculated Stress Values	Specific calculated stress values for ISFSI components and sub-components are deleted from the USAR as unnecessary details. The allowable stress values and the statement that the calculated stresses are within the allowables are left in.	None
6	New Design Temperature Value	A new higher design temperature value of 460 °F is used to be consistent with the SER and USAR. The affected structural analyses are revised.	Ref. 6

Specific changes to the USAR are being made in Chapters 3, 4 and 8. A log of the changes being made, along with category to which each change belongs, is shown in Attachment 1. Where applicable, the ITR discrepancy number is also listed against the change.

ACTIVITY: ES199800312	50.59 Log No.: N/A	72.48 Log No.: SE00158

ISFSI – Disposition of ITR Issues and Other USAR Changes

Background

The Independent Spent Fuel Storage Installation (ISFSI) at the Calvert Cliffs Nuclear Power Plant (CCNPP) utilizes the Nutech Horizontal Modular Storage (NUHOMS)-24P dry storage system. The system consists of concrete horizontal storage modules (HSMs), which provide passive storage for spent fuel assemblies that are placed within Dry Storage Canisters (DSCs). Twenty-four spent fuel assemblies are loaded into each DSC. Each DSC contains an outer leak-tight shell and an internal basket assembly. The outer shell provides the structural strength, shielding, and a leak-tight chamber for containing helium. The helium provides an inert atmosphere within the DSC. The internal basket assembly includes twenty-four stainless steel guide sleeves (one for each spent fuel assembly), nine perforated carbon or stainless steel spacer discs, and four stainless steel support rods. The nine spacer discs are spaced out along the length of the DSC at locations that approximately coincide with the spent fuel assembly's eight spacer grids and the single lower retention grid. The spacer discs are not structurally attached to the DSC shell walls or inner cover plates. The guide sleeves traverse the length of the DSC cavity through openings in the spacer discs. The support rods are used to maintain the spacer disc locations. The support rods traverse the length of the DSC cavity through the spacer discs, and are structurally welded to the spacer discs.

The fuel assemblies are loaded into the DSC, which is placed inside a Transfer Cask (TC), within the fuel handling building. The TC is then transported to the storage facility.

CCNPP utilizes the Combustion Engineering 14X14 fuel assemblies. Each of the fuel assemblies is approximately 157 inches in length and approximately eight by eight inches square in cross section.

Analyses / Justifications

As stated in the Proposed Activity above, the changes being made accomplish three objectives. The analyses and justification for the respective changes are as follows.

1. Resolution of ITR Discrepancies

Baltimore Gas & Electric (BG&E) identified potential discrepancies related to ISFSI during efforts associated with NRC's Confirmatory Action Letter (CAL), dated July 7, 1995, on Docket No. 72-1004. The discrepancies were related to pressure testing, design input values, and other code commitments. In order to establish the scope and safety significance of these discrepancies, BG&E contracted Transnuclear West (TN-W) to conduct an Independent Technical Review (ITR) [Ref. 3]. The scope of the ITR was to compile all safety significant licensing commitments and then verify that those commitments have been satisfied in the areas of design, fabrication, testing, surveillance, installation, and operations. The scope also included identifying discrepancies or failures to comply with the commitments, and defining actions to resolve the discrepancies such that, after completion of the actions, license compliance will be achieved with all safety significant commitments.

The ITR by Transnuclear West identified 43 discrepancies, and recommended actions to resolve them. These are duplicated in Attachment 2. Thirty-three of the discrepancies, which are marked with an asterisk in Attachment 2, are included in the scope of this safety evaluation.

BG&E contracted Hopper & Associates to evaluate the actions recommended by TN-W, and assist in closing them. The evaluation of the discrepancies and recommendations by Hopper & Associates are documented in References 4 and 5. Hopper & Associates also prepared documents that were necessary to resolve the discrepancies. These documents consisted of two calculations [Refs. 6 and 7], and changes to the USAR, which are included in Ref. 2.

Reference 6 is a re-analysis of the structural integrity of the Dry Shielded Canister (DSC). It resolves the discrepancies that were identified based on the previous design basis calculation. A complete discussion of how each discrepancy was resolved is included in References 4 and 5. Results of this calculation were incorporated into the USAR, and evaluated for safety, in accordance with 10CFR72.48, in References 8 and 9.

Similarly, Reference 7 is a re-analysis of the structural integrity of the Transfer Cask (TC). It resolves the discrepancies that were identified based on the previous design basis calculation. A complete discussion of how each discrepancy was resolved is included in References 4 and 5. Results of this calculation were incorporated into the USAR, and evaluated for safety, in accordance with 10CFR72.48, in Reference 10.

Another major change to the USAR is the addition of load combination Tables. Load combinations are currently cited in the USAR through references to the Topical Report [Ref. 12]. The new Tables being added now are based on the TR Tables, but are modified to incorporate site-specific parameters. For example, the load combinations for the flooding scenarios are being deleted, because the Calvert Cliffs ISFSI site is not susceptible to flooding.

2. DSC Design Temperature

The DSC design temperature used in the earlier analyses was 400 °F, instead of the correct value of 460 °F, as listed in USAR Table 3.6-3. The design temperature affects the stress allowables, and the pressures that would be developed within the DSC in the abnormal scenarios. Reference 6 documents the DSC analysis based on the correct design temperature. The new stress allowables were determined to be higher than the calculated stresses, and the pressures developed were also determined to be acceptable for the stresses that they would induce and were lower than the relief valve settings. Therefore, the DSC design is safe even with the new higher design temperature.

3. Deletion of Specific Calculation Results

The actual calculated results of DSC and TC sub-component stresses, temperatures and pressures were reported in the USAR Tables and text, and in some cases the calculational models were depicted in Figures. Such level of detail is not required in the USAR, and is being deleted. The information left in the USAR consists of the allowable parameters, and assessments that the calculated parameters are below the allowables.

NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage Facilities, [Ref. 11] was consulted to determine if the information being deleted is expected to be provided in the USAR. The accident analysis section of the SRP expects an "evaluation of calculated stress intensity level against the allowable stress intensity level…" Therefore, it is acceptable to include an assessment of calculated stresses and other parameters against the allowables, and delete the actual calculated parameters.

The deletion of these details is also supported by NEI 98-03, Revision 1 "Guidelines for Updating Final Safety Analysis Reports", which received NRC endorsement through Regulatory Guide 1.181. Specifically, Appendix A, Page 4 of the NEI document states:

"The following types of excessively detailed textural information may be removed from UFSARs, except as indicated by applicable regulatory guidance or NRC Safety Evaluation Reports:

Criterion 4 – Analytical information, e.g., detailed calculations, that are not important to providing an understanding of the safety analysis methodology, input assumptions and results, and/or compliance with relevant regulatory and industry standards."

The multitudes of changes being made to the USAR are listed in Attachment 1, and are categorized into six categories as listed above in the Proposed Activity. Analyses that support the changes are also referenced in the same place.

ACTIVITY: ES199800312	**50.59 Log No.: N/A**	**72.48 Log No.: SE00158**

ISFSI – Disposition of ITR Issues and Other USAR Changes

Reason for Activity:

This activity incorporates into the USAR the resolution of ITR discrepancies. In addition, it incorporates the results of DSC structural analysis at the correct higher design temperature, and deletes unnecessary details from the USAR. The ITR was performed in connection with the CAL issued by the NRC. The resolution of the ITR discrepancies provides assurance that the licensing commitments made in connection with the ISFSI would be met.

The incorporation of DSC analysis results, performed at the correct design temperature, is necessary to correct the earlier use of a lower design temperature.

The deletion of unnecessary information from the USAR is important, because otherwise any minor changes to dimensions or parameters, which may provide inconsequential changes in calculated results, would require revisions to the USAR.

Function(s) of affected SSCs:

The affected SSCs are the fuel assemblies, DSCs, TCs, and HSMs.

Fuel Assembly

The fuel assembly consists of 176 fuel and poison rods, 5 guide tubes, 5 guide tube sleeves, 8 fuel rod spacer grids, upper and lower end fittings, lower retention grid, and a hold-down device. The guide tubes, spacer grids, and end fittings form the structural frame of the assembly. The fuel rod spacer grids maintain the fuel rod pitch over the length of the assembly. The grid provides positive side restraint to the fuel rod but only frictional restraint axially. The spacer grids are the widest part on a fuel assembly. The four outer guide tubes are mechanically attached to the end fittings and the spacer grids are welded to all five guide tubes. The upper end fitting attaches to the guide tubes to serve as an aligning and lifting device for each fuel assembly. The fuel rods are held together in the assembly at a pitch of 0.58 inch, which helps to control the criticality. The rods consist of enriched uranium fuel pellets stacked within a Zircalloy cladding tube. The cladding is the first barrier that prevents the fission products from escaping to the outside.

Dry Storage Canister

The DSC is classified as important-to-safety per 10 CFR 72. It consists of the outer canister and the internal basket assembly. The sub-components of the internal basket assembly include the Spacer Discs, Support Rods, and Guide Sleeves. The internal basket assembly components are not attached structurally to the outer canister.

The DSC provides containment, shielding, criticality control, configuration control related to fuel retrievability, structural support, and thermal safety functions during loading operations, transfer operations, and storage. It is designed to remain intact under all accident conditions identified in the ISFSI USAR with no loss of function. Specific design functions of the DSC include the following:

1. Confinement - The DSC design provides mechanical confinement of the stored fuel assemblies to prevent the dispersion of particulate or gaseous radionuclides from the fuel. The primary function of the DSC is to provide confinement of the spent nuclear fuel. This is achieved by the stainless steel shell and two inner cover plates (top and bottom ends) which are welded to the shell assembly. There are also outer cover plates (top and bottom) to further assure containment integrity. The DSC confinement boundary is designed also to retain helium cover gas around the fuel in order to prevent corrosion of the fuel cladding and formation of expansive oxides in the fuel during storage.
2. Criticality Control - The DSC design provides for sub-criticality during the wet loading, DSC drying, and interim storage operations. This is accomplished by a combination of mechanical separation of

the fuel assemblies by the internal basket assembly and neutron absorption in the steel guide sleeve material.

3. Fuel Support and Configuration Control - The DSC internal basket assembly provides support for the spent fuel assemblies during normal operations. The DSC also provides configuration control related to post accident recovery of spent nuclear fuel. The DSC is designed so that the worst-case postulated accidents, including a cask drop, will not result in deformation of the Internal Basket Assembly or the DSC shell to such a degree that retrieval of intact fuel assemblies is not assured.
4. Shielding - The DSC materials provide gamma radiation shielding. The DSC provides gamma shielding at its ends by the use of lead shield plugs. These provide ALARA dose rates at the top of the canister during drying and sealing operations and at the bottom for minimizing dose rates during DSC loading into the Horizontal Storage Module (HSM) and at the HSM door during storage.
5. Thermal - Decay heat is removed by thermal radiation and conduction from the DSC to the TC, and by thermal radiation and conduction and convection from the DSC to the HSM. The DSC maintains the helium cover gas, which is required for corrosion control. This cover gas improves the thermal performance of the DSC.

The functions of the internal basket assembly components are as follows:

6. Guide Sleeves – The guide sleeves establish storage compartments for 24 spent fuel assemblies within the DSC. The tops of the guide sleeves are flared to assist fuel-handling operators in guiding the spent fuel assemblies into the sleeves.
7. Spacer Discs – The spacer discs work together with the guide sleeves to maintain geometric separation of the fuel assemblies. The spacer discs support the weight of the guide sleeves, support rods and the spent nuclear fuel when the DSC is in a horizontal orientation.
8. Support Rods – The support rods maintain the spacer disk locations along the length of the DSC. They carry the weight of the guide sleeves and the spacer discs when the DSC is in a vertical orientation.

Transfer Cask

The TC is important to safety because it protects the spent fuel container during handling. The TC is also considered safety-related since dropping a loaded TC, which weighs about 95 tons, has the potential for creating unanalyzed accident situations in the power plant. The TC is a cylindrical vessel with a bottom end closure assembly and a bolted top cover plate. Its cylindrical walls are formed from three concentric steel shells, the middle of which is the structural shell. Lead is poured between the inner and the middle shells to provide gamma shielding, and a solid neutron shielding material is poured between the middle and the outer shells. Two upper trunnions are located near the top of the cask, and are used for lifting. Two lower trunnions serve as the axis of rotation of the TC, and provide support.

The solid neutron shielding material used in the cask outer annulus, and top and bottom covers produces water vapor and a small quantity of non-condensable gases when heated to above 212 °F. The off gassing produces an internal pressure, which increases with temperature.

Horizontal Storage Module

The HSM is considered important to safety because it protects the DSC during storage. It consists of a concrete structure that provides shielding and support for the DSCs.

ACTIVITY: ES199800312	50.59 Log No.: N/A	72.48 Log No.: SE00158

ISFSI – Disposition of ITR Issues and Other USAR Changes

ISFSI USAR Revision No.: 9

ISFSI USAR Sections Reviewed:

Because of the large scope of changes being made to the USAR, almost the entire USAR was reviewed, except the site-related Sections. In particular, the main chapters reviewed in their entirety were 1, 3, 4, 5, 7, and 8.

Tech Spec Bases Amendment/Rev No.: 2

Calvert Cliffs Independent Spent Fuel Storage Installation Technical Specifications, Appendix A to Materials License No. SNM-2505, Amendment 2, June 30, 2000

Tech Spec Bases Reviewed:

2.1	Fuel to be Stored at ISFSI
2.2	Dry Shielded Canister (DSC)
2.3	Transfer Cask (TC)
2.4	Horizontal Storage Module (HSM)
3/4.1	Fuel to be Stored at ISFSI
5.0	Design Features

Complete for 50.59 and 72.48:

1. **The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.**

☐ YES ☒ NO **May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Probability of Malfunction:

The proposed activity consists of making changes to the ISFSI USAR. The changes are being made to resolve discrepancies in the USAR identified during the ITR, correct the DSC design temperature value to 460 °F, and delete the specific calculated values of individual component weights, stresses and temperatures. The proposed activity does not involve any hardware changes. Analyses that form the bases of the USAR changes have shown that none of the components of ISFSI are adversely impacted. The calculated stresses are below the allowables. The information being deleted from the USAR does not establish the safety of the related components.

Therefore, the probability of malfunction of equipment important to safety will not be increased because of the proposed changes.

☐ YES ☒ NO **May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Consequences of Malfunction:

The malfunctions to be considered are of the ISFSI important-to-safety components listed above.

The consequences of failure of the fuel assemblies, DSC, TC, and the HSM are all related to the release of radioactivity into the atmosphere or the dose to operators or the public. The fuel characteristics, such as the decay heat level and radiation dose criteria applicable to ISFSI components, are not altered. Also, the shielding and containment properties of the DSC and TC are not compromised. Therefore, the consequences of failure of any of the above equipment will not be impacted by this activity.

☐ YES ☒ NO **May the probability of occurrence of an accident previously evaluated in the SAR be increased?**

Probability of Accident:

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. They consist of loss of shielding, external missiles, earthquake, flood, cask drop, lightning, blockage of air inlets and outlets, DSC leakage, DSC overpressurization, and forest fire. However, the probability of occurrence of cask drop, or any other accident, is not increased by the changes being made to the USAR.

There is no change to the design or operation of the NUHOMS system caused by this activity. This activity does not modify the external configuration of the DSC envelope. The interface between the DSC and the HSM during ISFSI operations and interim storage of the DSC remains unaffected. Therefore, the probability of occurrence of an accident involving loss of HSM air outlet shielding, or blockage of HSM air inlets and outlets will not increase.

Pressurization of the DSC due to fuel cladding failure is an accident scenario identified in USAR Section 8.2.9. The limiting DSC pressurization accident event is a rupture of fuel cladding together with blockage of the HSM vents. This activity does not compromise the fuel cladding, or the fuel rod integrity.

DSC leakage is an accident scenario described in USAR Section 8.2.8. The USAR indicates that there are no credible events that would initiate this type of accident. As stated in the preceding paragraphs, the probability of an accident that would lead to cladding failure is not increased by this activity. This activity does not affect the design of the DSC pressure boundary and therefore does not increase the probability of DSC leakage.

☐ YES ☒ NO **May the consequences of an accident previously evaluated in the SAR be increased?**

Consequences of Accident:

The proposed activity, namely the USAR changes related to the resolution of ITR discrepancies and other items, is related to several accidents, particularly the cask drop accident, as stated above.

The consequences of the cask drop accident on the TC and DSC were evaluated in References 6 through 10. The impact on critical safety functions was evaluated in the same references. The fuel assembly evaluation is the subject of another safety evaluation (SE00154), which is under preparation.

Other changes being made to the USAR consist of revision to the load combination tables, incorporation of DSC analysis results that was performed with the correct design temperature, and the deletion of specific calculated sub-component stresses and temperatures. None of these activities affects the fuel cladding or the fuel rod integrity. Information important to safety is not being deleted. The information being deleted meets the NEI 98-03 criteria for deletion.

Therefore, consequences of an accident previously evaluated in the SAR will not be increased.

2. **The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.**

☐ YES ☒ NO **May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?**

Possibility of New Malfunction:

The proposed activity makes changes to the USAR to incorporate the resolution of ITR discrepancies and other items. None of the changes impact the environment, functioning, or the procedures related to the equipment important to safety. The information being deleted from the USAR is not important to safety. Therefore, there is no possibility created of a new malfunction in any of the important-to-safety ISFSI components.

☐ YES ☒ NO **May the possibility of an accident of a different type than any previously evaluated in the SAR be created?**

ACTIVITY: ES199800312	50.59 Log No.: N/A	72.48 Log No.: SE00158

ISFSI – Disposition of ITR Issues and Other USAR Changes

Possibility of New Accident:

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the USAR, and have been discussed previously. Evaluation of the proposed changes to the USAR showed that the important-to-safety components of ISFSI would maintain their safety functions. Since there is no change to the design or operation of the NUHOMS system caused by this activity, the possibility of an accident of a different type than any previously evaluated in the SAR would not be created.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ YES ☒ NO **Will the margin of safety as defined in the basis for any Technical Specification be reduced?**

Tech Spec Bases: 2.3

Discussion of why the margin of safety is not reduced:

The margin of safety is defined as the range of values between the acceptance limit reviewed and approved by the NRC as part of the licensing basis and the failure point [Ref. 14]. USAR Section 3.2.5 defines the acceptance criteria for ISFSI components, none of which would be exceeded. The use of the correct design temperature for DSC does reduce the allowable stresses, but the new values are still below the code allowables and hence do not represent a change in the licensing basis. The allowable stresses are not being deleted from the USAR. Therefore, the margin of safety would not be reduced.

Complete for 72.48:

☐ YES ☒ NO **Will the proposed activity involve a significant increase in occupational dose?**

A significant increase in occupational dose:

The radiation protection design and operation of the NUHOMS-24P dry cask storage system would not be changed by this proposed activity. The supporting analyses showed that the ISFSI equipment would maintain the radioactivity confinement boundary, and maintain the criticality control. Retrievability of fuel following a drop would still be assured. Occupational dose associated with post DSC accident recovery is not addressed in the Calvert Cliffs ISFSI USAR, however, the occupational dose for fuel retrieval, following the unlikely cask drop accident, is expected to be minimized through the use of special procedures and precautions. Because none of these attributes would be changed, the occupational doses summarized in USAR Table 7.4-1 would not be affected by this activity.

☐ YES ☒ NO **Will the proposed activity involve a significant unreviewed environmental impact?**

A significant unreviewed environmental impact:

The NUHOMS-24P dry cask storage system confinement and radiological shielding functions would not be reduced by this activity.

This activity would not affect any area of the plant site previously undisturbed for the ISFSI, and would not cause any reason for revision to the ISFSI Updated Environmental Report. This activity would not affect the environmental conditions associated with the ISFSI. Therefore, this activity would not involve an unreviewed environmental impact.

References:

1. Calvert Cliffs Independent Spent Fuel Storage Installation USAR, Rev. 9
2. CCNPP ISFSI USAR Change Request, UCR-00194
3. Transnuclear West Report No. 0113-92003-R-001, Independent Technical Review of Baltimore Gas & Electric Calvert Cliffs NUHOMS Project, Rev. 1, 9/1/98
4. Hopper and Associates Letter No. HABGE-06/99-0789 with Attachment, Resolution of Transnuclear West Independent Technical Review Comments for the BGE Calvert Cliffs NUHOMS Project, 6/2/99
5. Hopper and Associates Letter No. HABGE-10/00-0908, ITR Issues Disposition Response, 10/17/2000
6. BGE Calculation CA04132, Rev. 0002, Nutech Horizontal Module System (NUHOMS) 24P ISFSI Dry Shielded Canister Structural Analysis for DSC Assemblies R001-R024
7. BGE Calculation CA04141, Rev. 0002, ISFSI Transfer Cask Structural Evaluation
8. BGE 10CFR72.48 Safety Evaluation, SE00146
9. BGE 10CFR72.48 Safety Evaluation, SE00148
10. BGE 10CFR72.48 Safety Evaluation, SE00157
11. NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage Facilities (Final Report), March, 2000
12. Topical Report for the NUTECH Horizontal Modular Storage (NUHOMS) System for Irradiated Nuclear Fuel, NUH-002, Rev. 2
13. Calvert Cliffs Updated Final Safety Analysis Report (UFSAR), Rev. 26
14. NEI 96-07, Rev. 0, Guidelines for 10 CFR 50.59 Safety Evaluations, 09/97
15. Calvert Cliffs ISFSI Updated Environmental Report, Rev. 1
16. Technical Specifications for Calvert Cliffs ISFSI, Amendment 2

ACTIVITY: ES199800312	50.59 Log No.: N/A	72.48 Log No.: SE00158

ISFSI – Disposition of ITR Issues and Other USAR Changes

Summary: (For NRC Report, provide a brief overview)

Proposed Activity:

The proposed activity consists of making changes to the ISFSI USAR. The changes are being made to (1) resolve discrepancies in the USAR identified during an Independent Technical Review (ITR), (2) correct the DSC design temperature value to 460 °F, and (3) delete the specific calculated values of individual component weights, stresses and temperatures. The proposed activity does not involve any hardware changes.

Reason for Activity:

This activity incorporates into the USAR the resolution of ITR discrepancies. In addition, it incorporates the results of DSC structural analysis at the correct higher design temperature, and deletes unnecessary details from the USAR. The ITR was performed in connection with the CAL issued by the NRC. The resolution of the ITR discrepancies provides assurance that the licensing commitments made in connection with the ISFSI would be met.

The deletion of unnecessary information from the USAR is important, because otherwise any minor changes to dimensions or parameters, which may provide inconsequential changes in calculated results, would require revisions to the USAR.

Activity Summary:

Several changes to the USAR are being made in chapters 3, 4, and 8. The changes are categorized as follows:

- Load combinations used in the structural analyses of ISFSI components have been re-defined using site specific information.
- The transfer cask structural integrity was re-analyzed.
- Discrepancies identified in the thermal/ pressure parameters are resolved.
- The DSC structural integrity was re-analyzed.
- Specific calculated stress values for ISFSI components and sub-components are deleted from the USAR as unnecessary details. The allowable stress values, and the statement that the calculated stresses are within the allowables, are left in. The deletion was justified, as such information was not listed as required in NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage Facilities.
- A new higher design temperature value of 460 °F is used to be consistent with the SER and USAR. The affected structural analyses are revised.

USQ Determination: This activity was evaluated against the criteria of 10CFR72.48(a)(2), such as the probability of occurrence or the consequences of an accident or the malfunction of equipment important to safety, and it was concluded that it does not involve an unreviewed safety question (USQ).

USAR Change Log

Change No.	USAR Section	USAR Page	Description	Change Category	ITR Discrep-ancy No.
1	3.2.5	3.2-3	Add text references to new tables of site-specific load combinations.	1	D102, D310
2	T3.2-2	New	Copy HSM load combinations table from the Topical Report (TR).	1	D102
3	T3.2-3	New	Copy TR Table 3.2-5a, and modify it to make it site-specific.	1	D102
4	T3.2-4	New	Copy TR Table 3.2-5b, and modify it to make it site-specific.	1	D102
5	T3.6-1	3.6-4	Re-define design loads for transfer cask upper trunnions.	2	D301
6	T3.6-3	3.6-8	Revise DSC internal pressure design parameter to 50 psig	3	D602
7	4.7	4.7-2	Revise the bottom neutron shield relief valve setting to 40 psig.	3	D306
8	8.1.1.1.A	8.1-1	Text is revised to delete specific calculated component weights.	5	-
9	8.1.1.2	8.1-2	Text is revised in accordance with the new analysis, and to delete stress calculation model figures and result tables.	4, 5	-
10	8.1.1.3	8.1-3	Text is revised to delete stress calculation model figures and result tables.	5	-
11	8.1.1.4	8.1-4	Text is revised to delete stress calculation result tables.	5	-
12	8.1.1.5	8.1-4, -5	Text is revised to delete stress calculation result tables.	5	-
13	8.1.1.9	8.1-6	Text is revised in accordance with the new analysis, and to delete stress calculation model figures and result tables.	2, 5	-
14	8.1.2.1.C	8.1-7	Text is revised to delete stress calculation result tables.	5	-
15	8.1.2.2	8.1-8	Text is revised to delete stress calculation result tables.	5	-
16	8.1.3.1	8.1-11	Text is revised to delete temperature calculation model figures and result tables.	5	-
17	8.1.3.2	8.1-11, -12	Text is revised to delete temperature calculation model figures and result tables.	5	-
18	8.1.3.3	8.1-12	Text is revised to delete temperature calculation result table.	5	-
19	T8.1-1	8.1-14	Delete table of calculated weights.	5	-
20	T8.1-2	8.1-15	Delete table of calculated stresses.	5	-
21	T8.1-3	8.1-16	Delete table of calculated stresses.	5	-

ACTIVITY: ES199800312	50.59 Log No.: N/A	72.48 Log No.: SE00158

ISFSI – Disposition of ITR Issues and Other USAR Changes **Attachment 1**

Change No.	USAR Section	USAR Page	Description	Change Category	ITR Discrep-ancy No.
22	T8.1-4	8.1-17	Delete table of calculated stresses.	5	-
23	T8.1-5	8.1-18	Delete table of calculated stresses.	5	-
24	T8.1-6	8.1-19	Delete table of calculated stresses.	5	-
25	T8.1-7	8.1-20	Delete table of calculated deflections	5	-
26	T8.1-8	8.1-21	Delete calculated stress values.	5	-
27	T8.1-9	8.1-22, -23	Delete table of calculated temperatures.	5	-
28	T8.1-10	8.1-24	Delete table of calculated stresses.	5	-
29	T8.1-11	8.1-25	Delete table of calculated temperatures.	5	-
30	T8.1-12	8.1-26	Delete table of calculated temperatures.	5	-
31	T8.1-13	8.1-27	Delete table of calculated temperatures.	5	-
32	T8.1-14	8.1-28	Delete table of calculated temperatures.	5	-
33	F8.1-1	?	Delete calculation model.	5	-
34	F8.1-2	?	Delete calculation model.	5	-
35	F8.1-3	?	Delete calculation model.	5	-
36	F8.1-4	?	Delete calculation results.	5	-
37	F8.1-5	?	Delete calculation model.	5	-
38	F8.1-6	?	Delete calculation model.	5	-
39	F8.1-7	?	Delete calculation model.	5	-
40	F8.1-8	?	Delete calculation model.	5	-
41	N/A	N/A	Not used.	N/A	N/A
42	F8.1-10	?	Delete calculation results.	5	-
43	F8.1-11	?	Delete calculation results.	5	-
44	F8.1-12	?	Delete calculation model.	5	-
45	F8.1-13	?	Delete calculation results.	5	-
46	F8.1-14	?	Delete calculation results.	5	-
47	F8.1-15	?	Delete calculation results.	5	-
48	F8.1-16	?	Delete calculation results.	5	-
49	8.2.2.2. A	8.2-2	Revise text to delete reference to results tables that are being deleted.	5	-
50	8.2.2.2. B	8.2-2	Revise text to reflect the revised analysis.	2	D302
51	8.2.2.2. C	8.2-3	Revise text to delete reference to results tables that are being deleted.	5	-
52	8.2.2.2. D	8.2-3, thru –5	Revise text to reflect the revised analysis.	2	D302
53	8.2.3.2	8.2-6, -7	Revise text to delete reference to results tables that are being deleted.	5	-
54	8.2.5.2	8.2-9, -10	Revise text to reflect new analyses, and references to the deleted result tables.	2, 4, 5	D309, D314
55	8.2.7.2	8.2-12	Revise text to delete reference to results tables that are being deleted.	3	-
56	8.2.9.2	8.2-19	Revise the DSC internal pressure value.	3	-

SAFETY EVALUATION FORM

ACTIVITY: ES199800312	50.59 Log No.: N/A	72.48 Log No.: SE00158

ISFSI – Disposition of ITR Issues and Other USAR Changes **Attachment 1**

Change No.	USAR Section	USAR Page	Description	Change Category	ITR Discrep-ancy No.
57	8.2.12	8.2-22	Revise text to delete references to deleted results tables.	5	-
58	T8.2-2	8.2-25	Delete calculation results table.	5	-
59	T8.2-3	8.2-26	Delete calculation results table.	5	-
60	T8.2-4	8.2-27	Delete calculation results table.	5	-
61	T8.2-5	8.2-28	Delete calculation results table.	5	-
62	T8.2-6	8.2-29	Delete calculation results table.	5	-
63	T8.2-7	8.2-30	Delete calculation results table.	5	-
64	T8.2-8	8.2-31	Delete calculation results from the table.	5	-
65	T8.2-9	8.2-32	Delete calculation results from the table.	5, 6	-
66	T8.2-10	8.2-33	Delete calculation results from the table.	5, 6	-
67	T8.2-11	8.2-34	Delete calculation results table.	5	-
68	T8.2-12	8.2-35	Delete calculation results from the table.	5	-
69	T8.2-13	8.2-36	Delete calculation results table.	5	-
70	T8.2-14	8.2-37	Delete calculation results from the table.	5	D305
71	T8.2-15	8.2-38	Delete calculation results from the table.	5	D305
72	8.2-16	8.2-39	Delete calculation results from the table.	5	D305

ATTACHMENT H

BGE ITR Discrepancy Resolution

Discrep. No.	Description	Recommended Action	Action to be Completed by
D101	Design specification references incorrect fuel burnup values	Revise design specification	BGE 2F199800014 MS 001 GRYCZKOWSKI
D102 ☆	Clarify structural analysis loading conditions evaluated	Revise SAR to reconcile differences between Topical Report (referenced by SAR) and Design Specification/Calculations	BGE 2F199800014 MS 002 REMENIUK
D301 ☆	Discrepancy between SAR and calculation loads for the cask upper trunnion	Revise SAR	BGE 2F199800014 MS 003 REMENIUK
D302 ☆	Tornado analysis results for wind stability provided in SAR but not included in design calculations	Add analysis to design calculations and revise SAR if necessary	BGE 2F199800014 MS 004 REMENIUK
D303 ☆	Stress intensities not calculated in accordance with ASME Code	Add reference to this discrepancy and to the confirmatory analysis in the cask structural calculation	BGE 2F199800014 MS 005 REMENIUK
D304	Design analysis could not provide a detailed dose profile at the location of the docking ring.	Unresolved item. Review available dose data and document adequacy of local dose with respect to SAR limit.	BGE 2F199800014 MS 006 BEALL
D305 ☆	Incorrect allowable stress intensities reported in SAR.	Revise SAR.	BGE 2F199800014 MS 007 REMENIUK
D306 ☆	Test pressure for bottom shield cavity was 45 psig, SAR specifies 50 psig.	Revise SAR and/or revise relief valve setpoint.	BGE 2F199800014 MS 008 REMENIUK
D307 ☆	Incorrect reference to computer run providing the basis for thermal stress intensity calculation for the neutron shield jacket support rings.	Reference this discrepancy and the confirmatory analysis in the calculation.	BGE 2F199800014 MS 009 REMENIUK
D308 ☆	Elastic thermal stress analysis indicates stresses above 3Sm, therefore an elasto-plastic analysis is required.	After further review this concern was found to be related to a local discontinuity which does not require analysis per code requirements.	No action 2F199800014 MS 010 REMENIUK
D309 ☆	Error in data used to represent crush reaction for the side drop analysis.	Reference this discrepancy and the confirmatory analysis in the cask structural analysis.	BGE 2F199800014 MS 011 REMENIUK
D310 ☆	Load cases specified in SAR are inconsistent with Topical Report.	Revise SAR.	BGE 2F199800014 MS 012 REMENIUK

Discrep. No.	Description	Recommended Action	Action to be Completed by
D311 ☆	Tornado missile analysis does not consider a downward impact load on the top of the cask.	Document in design calcs that this load condition is not governing.	BGE 2F19980014 NS 013 REMENIUK
D312 ☆	Cask bottom plate stresses were calculated at the center of the plate only. Higher stresses at the fixed boundary were not evaluated.	Update analysis and SAR. Impact evaluated and found to be acceptable.	BGE 2F19980014 NS 014 REMENIUK
D313	Technical specification 10.3.2.6 requires surface swipes on inside but not exterior of cask. Swipes of external surface are not taken. This is not consistent with 49CFR173.443(d).	Revise SAR to clarify that CFR requirement is not appropriate due to ALARA concerns and is not necessary based on cask handling procedures.	BGE 2F19980014 NS 015 BEALL
D314 ☆	Cask corner drop stress summary reports equivalent stresses not stress intensities.	Update cask calc. Impact evaluated and found to be acceptable.	BGE 2F19980014 NS 016 REMENIUK
D401 ☆	ANSI N14.6 stress analysis requirements not rigorously followed.	Review indicates acceptability of design. Incorporate into cask structural analysis.	BGE 2F19980014 NS 017 REMENIUK
D501 ☆	Assumptions used in missile load on analysis of HSMs appears to be inconsistent with respect to target ductility assumptions versus the effect of design methods on material ductility.	Incorporate a discussion of this issue into the HSM calculations to substantiate the assumptions used.	BGE 2F19980014 NS 018 REMENIUK
D502 ☆	Inconsistency in dimensions between drawings and specifications.	Review and resolve the inconsistency by revising the appropriate documents.	BGE 2F19980014 NS 019 REMENIUK
D503	Documentation of the acceptability of DSC support steel lubricant with respect to environmental qualification (temperature and dose) was not included in design basis analyses.	Incorporate a discussion of the vendor qualification of the lubricant for the environmental conditions in analyses or specifications.	BGE 2F19980014 NS 020 BEALL
D504 ☆	The method used for summation of load cases for the HSM SAR analysis was not appropriate.	Revise SAR to reflect the appropriate method of load combination. No impact on acceptability of results.	BGE 2F19980014 NS 021 REMENIUK
D601 ☆	The SAR/SER maximum operating temperatures reported for determination of stress allowables is not consistent with the design analyses.	Revise the SAR to reflect the appropriate temperatures and allowables. No impact on acceptability of results.	BGE 2F19980014 NS 022 REMENIUK
D602	DSC internal pressure for accident	The design conditions used	BGE

Discrep. No.	Description	Recommended Action	Action to be Completed by
D602 ☆	conditions reported in the SAR is not consistent with the analysis.	envelope the analysis value. Revise the SAR to reflect the correct values.	2F19980014 MS023 REMENIUK
D603 ☆	*Summation of stresses for the* various load cases may not be conservative.	*Revise design analyses to* document the basis for acceptability of the method used. See confirmatory analysis for review *and* acceptance of the approach used.	BGE 2F19980014 MS 024 REMENIUK
D604 ☆	DSC calculations used ASME section NB criteria where NG criteria may have been more appropriate.	Revise analysis to document the basis and acceptability of the method used.	BGE 2F19980014 MS025 REMENIUK
D605 ☆	Stresses above 3Sm were determined without application of plastic analysis acceptance criteria.	Confirmatory analysis determined that stresses are within allowables. Revise SAR to reflect the results of the confirmatory analysis.	BGE 2F19980014 MS026 REMENIUK
D606	DSC thermal analysis model used for heat transfer *in HSM may* not be accurate.	Considered to be an insignificant impact. Incorporate basis for acceptability of the thermal model into thermal analysis.	No action required 2F19980014 MS027 MASSOUD
D607 ☆	Finite element model used for spacer *disc may not provide* accurate results.	Confirmatory analysis indicates *that results are acceptable.* Incorporate a discussion of the confirmatory analysis in the design calculation.	BGE 2F19980014 MS 028 REMENIUK
D608 ☆	*Thermal modeling of DSC in* transfer cask may not be appropriate.	*After further review it was* determined that the model used was sufficiently accurate. Incorporate a discussion of the *acceptability of the model* into the design calculation.	No action required 2F19980014 MS 029 REMENIUK
D609 ☆	ANSYS nodal temperature input was incorrect.	Confirmatory analysis indicates that the analysis results are acceptable when the error is corrected. Revise design calculation to reference confirmatory analysis or incorporate a discussion of the impact of the error.	BGE 2F19980014 MS030 REMENIUK
D610	Thermal analysis incorrectly references 8 year old *fuel* as basis for evaluation. The correct heat load associated with 10 year old	Revise analysis to correct the *erroneous statement.*	BGE 2F19980014 MS031 BEALL

Discrep. No.	Description	Recommended Action	Action to be Completed by
	fuel is used in the analysis.		
D611 ☆	Structural analysis of spacer disc in basket does not model interaction between spacer discs and the potential impact on ou-of-plane bending.	Confirmatory analysis indicates that code allowables are met when a more appropriate model is used. Revise design analysis to reference confirmatory analysis.	BGE 2F199800014 MS 032 REMENIUK
D612 ☆	Plastic analysis modulus of elasticity for plastic region not consistent with ANSYS/Code recommendations.	Confirmatory analysis performed confirming adequacy with respect to the appropriate code allowables. Revise design analysis to reference/incorporate the confirmatory analysis.	BGE 2F199800014 MS 033 REMENIUK
D613 ☆	DSC shell local bending circumferential stress does not using the bounding DSC weight of 80,000 lbs. The associated stress is calculated as a stress intensity without justification.	The resulting stresses are not controlling and therefore the error does not impact the bounding analyses. Revise the design analysis to justify the approach used or to revise to incorporate the bounding weight.	BGE 2F199800014 MS 034 REMENIUK
D614 ☆	Modeling of support ring welds is not accurate.	A confirmatory analysis using more a more appropriate model supports the calculation results. Incorporate a discussion of the confirmatory analysis, or reference to it, in the design calculation.	BGE 2F199800014 MS 035 REMENIUK
D615 ☆	Modeling of DSC shell/lead shield plug interface is not substantiated.	Further review indicates that the model used was appropriate.	No action required 2F199800014 MS 036 REMENIUK
D616 ☆	Basis for analysis of support *rod/discs with respect to NG2330* not found.	Subsequent review has found the appropriate basis/analysis.	No action required 2F199800014 MS 037 REMENIUK
D617 ☆	Fabrication specification, NUH-03-107 does not require proof pressure testing.	Addressed in response to confirmatory action letter. Results of this review have been incorporated in Revision 6 of NUH-03-107 and associated issued ECNs 95-0021, 95-0037 and 96-0067.	No action required 2F199800014 MS 038 REMENIUK
D618 ☆	Inconsistent reference to applicable ASME code section for the DSC basket (NG vs. NB).	Review and update SAR as necessary to make it consistent with SER. No impact on analysis.	BGE 2F199800014 MS 039 REMENIUK
D619	ORIGEN code calculations of source term used incorrect input data.	The erroneous input data was not used in the analysis. Revise the calculation to reference the correct data.	No action required 2F199800014 MS 040 GRYCZKOWSKI

SAFETY EVALUATION FORM

ACTIVITY: ES199800312	50.59 Log No.: N/A	72.48 Log No.: SE00158

ISFSI – Disposition of ITR Issues and Other USAR Changes **Attachment 2**

Discrep. No.	Description	Recommended Action	Action to be Completed by
D620	Secondary gamma sources were not appropriately modeled in dose analysis model.	A review of the actual data used and its generic basis which envelopes BGE Calvert Cliffs fuel indicate that the existing analysis is adequate.	No action required 2F19980014 NS041 GRYCZKOWSKI
D621	Neutron and gamma dose rates were benchmarked against a fuel different than the design basis fuel.	Further review indicates that the benchmarks were bounding for the design basis fuel.	No action required 2F19980014 NS042 GRY...SKI
D622	DSC bottom end closure welds were tested using a soap bubble test instead of a helium leak test.	A CAR has been issued to review and address this issue. Further testing may be required to document license compliance. Testing of unloaded DSCs indicates that the fabricated components meet the helium leak test criteria.	BGE to evaluate TNW basis for acceptability provided for this discrepancy. BGE to assess need for further testing. 2F19980014 NS043 BEALL

ATTACHMENT 3, SAFETY EVALUATION FORM

Page 1 of 5

ACTIVITY: ES200001036-000 | **50.59 LOG NO.:** | **72.48 LOG NO.: SE00159**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety questions (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions?
☐ YES ☒ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: B. H. Scott
PRINTED NAME AND SIGNATURE
Department: 42-01-03
Date: 11/3/00

☒ YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp Ind.:	Resp. Ind.:	Resp. Ind.:
R. H. Beall **PRINTED NAME**	**R. O. Hardies** **PRINTED NAME**	**PRINTED NAME**
SIGNATURE	**SIGNATURE**	**SIGNATURE**
Work Group: NFM	Work Group: MEIU	Work Group:
Date: 11/6/00	Date: 11/6/00	Date:

Approved ☑ Disapproved ☐
Signature: **M. A. BLACKWELL**
INDEPENDENT REVIEWER
Date: 11/6/00

Approved ☑ Disapproved ☐
Signature **W.E. KEMPER**
GS-DES, GS-TSES, OR PE-PDSU
Date: 11-6-00

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 00-095 Date: 11/6/00

Recommend Approval ☑ Recommend Disapproval ☐ Signature
POSRC CHAIRMAN
Date: 11/6/00

Approval ☑ Disapproved ☐ Signature
PLANT GENERAL MANAGER
Date: 11/6/00

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ YES ☒ NO

Signature:
OSSRC SES Chairman
Date: 01/12/01

If yes, OSSRC Meeting No.

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES200001036-000 | **50.59 LOG NO.:** | **72.48 LOG NO.: SE00159**

Proposed Activity: Allow use of Dry Shielded Canister (DSC) BGE-24P-W042 without the four stainless steel plugs installed in the top spacer disc of the basket assembly.

Reason for Activity: The ISFSI USAR states that all DSC structural components are fabricated from type 304 stainless steel, except the spacer disks and support rods may be fabricated from aluminum coated carbon steel. BGE requested an alternative material for the spacer disks and support rods to reduce fabrication costs. The top spacer disc of DSC BGE-24P-W042 is made of carbon steel and coated with a flame spray aluminum to provide the necessary corrosion resistance. There are four threaded holes in the top spacer disk used during fabrication of the canister. These four holes are not sprayed with the aluminum coating. To protect the carbon steel surfaces in these holes, stainless steel plugs are inserted prior to shipping the DSC. The stainless steel plugs for DSC BGE-24P-W042 were mistakenly removed at CCNPP during preparations for loading the DSC. The DSC has been filled and the top shield plug assembly and top cover plate have been welded in place.

Function(s) of affected SSC. NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are three major components of the NUHOMS-24P system that are addressed in this safety evaluation. Those three components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); and 3) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those three components.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2002. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. (DSC BGE-24P-W042 is aluminum coated carbon steel.) The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and act as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

SAR Revision No.: 9
SAR Sections Reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.4, 3.6, 4.2, 5.1, 7.4, 8.1, and 8.2.

Tech Spec Bases Rev. No. Amendment 2
Tech Spec Bases Reviewed: All

Complete for 50.59 and 72.48

ACTIVITY: ES200001036-000 | **50.59 LOG NO.:** | **72.48 LOG NO.: SE00159**

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ YES ☒ NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the USAR allowing the DSC spacer disks and support rods to be fabricated from type 304 stainless steel or aluminum coated carbon steel. The use of aluminum coated carbon steel was evaluated by Pacific Nuclear Fuel Services in 1991 via vendor calculation no. BGE001.0216 (Carbon Steel DSC Basket Assembly). The calculation evaluated the DSC for allowable stresses, ductility, and corrosion resistance.

Once the DSC has been loaded, the canister is evacuated and backfilled with helium. The EPRI NMAC "Boric Acid Corrosion Guidebook" states that the corrosion rate of carbon steel in the presence of dry boric acid crystals is essentially zero. Similarly, in deaerated solutions, the corrosion rate is less than 0.001" per year. Since the canister is initially evacuated, any contained water will be evaporated and removed. The helium backfill provides an inert long term environment. Therefore, a corrosion rate of <0.001" per year is appropriate. This will represent a corrosion depth of 0.050" over the 50 year design life of the canister, limited to the vicinity of the threaded holes. The plate was exposed to borated water in the spent fuel pool for approximately forty four hours during fuel loading. The EPRI Guidebook estimates a corrosion rate of <0.008" per year, i.e., <0.00005" corrosion for a 48 hour period, for boric acid concentrations and temperatures typical of the spent fuel pool. Therefore, corrosion of the exposed surface during this immersion period was negligible and limited to the surfaces in the threaded holes. The minimal corrosion that may be experienced during canister loading and over the 50 year design life will not adversely impact the structural integrity of the basket assembly under any of the design loading conditions. The original evaluation of the carbon steel plates concluded that there is sufficient margin in the calculated stresses to ensure that the corrosion resulting from a lengthy exposure of unprotected carbon steel to the pool environment would not jeopardize the safe operation of the DSC for any of its design basis functions. The strength of carbon steel for structural support of the stored spent fuel exceeds that of the stainless steel.

☐ YES ☒ NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the USAR allowing the DSC spacer disks and support rods to be fabricated from type 304 stainless steel or aluminum coated carbon steel. Likewise there are no malfunctions of the DSC described or evaluated in the USAR which are affected by the minimal corrosion potentially resulting from the omission of the four plugs in the upper spacer disc. As such, there are no consequences to consider.

☐ YES ☒ NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of this proposed activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The accident analysis included the use of either type 304 stainless steel or aluminum coated carbon steel spacer disks and support rods. The minimal corrosion expected in the vicinity of the four missing plugs will not negatively impact the results of the structural analysis.

☐ YES ☒ NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident: The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the use of either material was considered in the analysis, and since any potential corrosion resulting from the omission of the four plugs in the top spacer disc will not adversely impact the structural integrity of the assembly, there will be no increase in the accident dose consequences already described in the USAR.

ACTIVITY: ES200001036-000 | **50.59 LOG NO.:** **72.48 LOG NO.: SE00159**

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created

☐ YES ☒ NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction: The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. One possible malfunction of the DSC which is not described or evaluated in the USAR is the corrosion of the DSC carbon steel spacer disks and support rods due to exposure to spent fuel pool environment of borated water. The material corrosion properties are only relevant during transfer of fuel to the DSC in the spent fuel pool since the storage atmosphere is made inert with Helium and there is no oxygen present to support corrosion of the carbon steel spacer disks and support rods. As discussed previously, the corrosion rates predicted for carbon steel in the spent fuel pool environment during DSC fuel loading will result in minimal corrosion during the 48 hours the spacer plate was exposed to pool water.

☐ YES ☒ NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident: The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity.

Complete for 50.59 and 72.48

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ YES ☒ NO Will the margin of safety as defined in the basis for any Technical Specifications be reduced?

Bases	Discussion of why the margin of safety is not reduced
None of the Technical Specifications nor the Bases are affected by this activity.	

Complete for 72.48

☐ YES ☒ NO Will the proposed activity involve a significant increase in occupation dose?

A significant increase in occupational dose: A significant increase in occupational dose will not occur as a result of this proposed activity. This activity will allow the use of DSC BGE-24P-W042 without the four plugs installed in the top spacer disc. The exposure of portions of the carbon steel spacer plate to borated pool water during fuel loading will result in negligible corrosion and will not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

☐ YES ☒ NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact: A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

The proposed activity is to allow use of Dry Shielded Canister (DSC) BGE-24P-W042 without the four stainless steel plugs installed in the top spacer disc of the basket assembly. The ISFSI USAR states that all DSC structural components are fabricated from type 304 stainless steel, except the spacer disks and support rods may be fabricated from aluminum coated carbon steel. BGE requested an alternative material for the spacer disks and support rods to reduce fabrication costs. The top spacer disc of DSC BGE-24P-W042 is made of carbon steel and coated with a flame spray aluminum to provide the necessary corrosion resistance. There are four threaded holes in the top spacer disk used during fabrication of the canister. These four holes are not sprayed with the aluminum coating. To protect the carbon steel surfaces in these holes, stainless steel plugs are inserted prior to shipping the DSC. The stainless steel plugs for DSC BGE-24P-W042 were mistakenly removed at CCNPP during preparations for loading the DSC. The DSC has been filled and the top seal plug assembly and top cover plate have been welded in place.

Minimal corrosion expected during the short period of immersion in the spent fuel pool. After the DSC is loaded, it is evacuated and backfilled with helium. This creates an inert environment inside the DSC which will minimize any

ACTIVITY: ES200001036-000	50.59 LOG NO.:	72.48 LOG NO.: SE00159

corrosion of the exposed area of the spacer disc. The corrosion predicted over the 50 year design life of the DSC will not adversely impact the structural integrity of the canister. After a thorough review, it has been concluded that the proposed activity:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, 10 CFR 50.59/10 CFR 72.48 EVALUATION FORM (Page 1 of 7)

Proposed Activity NO: ES200101042	**50.59 Log No.:**	
	72.48 Log No.:	**SE00160**

Does the *Proposed Activity*:

☐ YES ☒ NO Require a License Amendment for a change to the Technical Specifications/License Conditions?

Based on this 10 CFR 50.59/10 CFR 72.48 evaluation, does the *Proposed Activity*:

☐ YES ☒ NO Require a License Amendment because it meets one (or more) of the eight (8) criteria of 10 CFR 50.59(c)(2)/10 CFR 72.48(c)(2)?

Prepared by: Kim I.R. Knippel/ Kim IR Knippel
PRINTED NAME AND SIGNATURE
Department: 4A-30-06 Date: 9/23/02

☒ YES ☐ NO Are cross-disciplinary concurrence reviews needed?

If 'YES', document completion of these reviews below:

Responsible Individual	**Responsible Individual**	**Responsible Individual**
Robert H. Beall	Getachew Tesfaye	
PRINT NAME	PRINT NAME	PRINT NAME
NFM/4A-30-03	NRM (4A-01)	
PRINT ORGANIZATION	PRINT ORGANIZATION	PRINT ORGANIZATION
/ 9/24/02	/ 09/24/02	/
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

Print/Signature: Gomk K. Gryczkowski
INDEPENDENT REVIEWER
Date: 9/23/2002

Print/Signature:
GS-DES, GS-TSES, or PE-PDSU
Date: 9/25/02

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 02-070 Date: 9/25/02

Recommend Approval ☑ Recommend Disapproval ☐

POSRC CHAIRMAN Date 9/25/02

Approved ☑ Disapproved ☐

PLANT GENERAL MANAGER Date 9/27/02

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? ☐ YES ☒ NO

Print/Signature: ANIL JULKA / Anil Julka
OSSRC SES Chairman
Date: 3-4-04

If 'YES', OSSRC Meeting No. ____________

Proposed Activity NO: ES200101042	**50.59 Log No.:**	
	72.48 Log No.:	**SE00160**

Proposed Activity (Description): 72.48 EVALUATION FOR THE ISFSI 24P TRANSFER CASK AND HSM DOSE RATE CALCULATIONS WITH MCNP

The Nuclear Engineering Unit has performed an Owner Acceptance Review (OAR) of the new source term, shielding and dose rate calculations that support the CCNPP ISFSI. These new calculations were required to correct all of the errors identified in several IR's (e.g., IR1-001-560, IR1-001-562, and IR1-001-563) and are documented in CA05923, CA05924, CA05925, and CA05926 which are contained in ES200101042. These CCNPP calculations document OAR of Duke Engineering & Services (DES) calculations CCNPP-DES-001/002/003/004 that will provide the new design basis calculations and models for the ISFSI 24P Dry Shielded Canister (DSC), Transfer Cask (TC), and Horizontal Storage Module (HSM). One additional in-house calculation has been performed (CA06058) to recalculate the ISFSI 24P 5 Phase Site Dose Rates. These new calculations require ISFSI SAR changes (UCR00317 & UCR00358), utilize new and improved methodology and have been performed with the Los Alamos National Laboratory Monte Carlo N-Particle Transport Code System MCNP4C. Prior to implementation of the new ISFSI design basis calculations and the ISFSI SAR changes (UCR00317 & UCR00358) a 72.48 evaluation will be required. These new calculations will supersede the existing calculations of record (that contain many errors and utilize less accurate deterministic methods) when the 72.48 Evaluation is completed, POSRC approved and documented in ES200101042. These new calculations result in < 10% decrease in the margin to all applicable dose and dose rate limits. Thus, replacing the existing calculations with these new ones will not require prior NRC approval. Utilization of MCNP itself as a new method/code in and of itself will not require prior NRC approval due to it's prior NRC SER acceptance for several Utilities and Private Companies that have used MCNP for criticality and shielding licensing analysis (Attachment 1). This conclusion is formally documented in ES200101042 and SE00160. ES200101042 also contains UCR00358. UCR00358 is an update to the ISFSI SAR chapters 1, 3, 7, and 8 to incorporate the new design bases dose rates obtained from ES200101042. UCR00358 also contains a new ISFSI SAR Figure 7.2-1 (Attachment 2 - "24P Design Basis Radiological Limit Curve") and some consolidation and elimination of redundent information in chapters 1, 3, 7 and 8. Adherance to both the heat generation rate and radiological source term limits will be proceduraly controlled via FH350 and CA05803.

Reason for Activity: IR200100638 Resolution and to enable the new Design Basis ISFSI shielding calculations with MCNP (contained in ES200101042) to supercede the original calculations.

Proposed Activity NO: ES200101042	**50.59 Log No.:**	
	72.48 Log No.:	**SE00160**

Affected SSC Design Functions:

The Dry Shielded Canister (DSC), Transfer Cask (TC), and Horizontal Storage Modules (HSM's) as shielding and storage devices for the spent fuel have not been affected by the new design basis calculations. However, the procedure for loading the spent fuel (FH-350) and the ISFSI SAR will be updated to reflect the new design basis dose rates obtained with the code MCNP4C (with the correction of deficiencies detailed in the Proposed Activity above). No physical changes to the DSC, TC or HSM are needed as a result of the new calculations. MCNP4C is simply an improved methodology that yields more accurate results (compared to the existing methodology of record). The radiological source term is the same as the original design basis and is calculated from ORIGEN2 for a CCNPP Westinghouse/Combustion Engineering 14X14 PWR fuel assembly with 3.40 w% U-235 enrichment and a burnup of 42 GWD/MTU at a decay time of eight (8) years. Industry standard and separate gamma and neutron axial flux shaping factors were utilized to develop the source term input for both the Cask and HSM models. Both Cask and HSM model dose rate results have been demonstrated conservative via benchmarks with actual CCNPP Cask and HSM loadings.

UFSAR/USAR Sections reviewed where relevant information was found:	Tech Spec Sections reviewed where relevant information was found:
1, 3, 5, 7, 8, and 9	2 and 3/4

Proposed Activity NO: ES200101042	**50.59 Log No.:**	
	72.48 Log No.:	**SE00160**

Does the *proposed activity*:

1. ☐ YES ☒ NO Result in more than a minimal increase in frequency of occurrence of an accident previously evaluated in the UFSAR/USAR?

Justification: This activity calculates the dose rates (and doses) associated with the CCNPP ISFSI Cask Loading and HSM facility by utilizing new calculational methodology and does not affect the frequency of occurrence of any accident.

2. ☐ YES ☒ NO Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component (SSC) important to safety previously evaluated in the UFSAR/USAR?

Justification: This activity calculates the dose rates (and doses) associated with the CCNPP ISFSI Cask Loading and HSM facility yielded by utilizing new calculational methodology and does not affect the likelihood of occurrence of a malfunction of any SSC

3. ☐ YES ☒ NO Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR/USAR?

Justification: There are only three accidents that have dose consequences. One is the Cask Drop Accident, the second is the Blockage of Air Inlets and Outlets and the third is the Dry Shielded Canister Leakage. The Cask Drop Accident has had its contact (1" from the side) dose rate increase from the currently documented value of 977 mrem/hr to 1126 mrem/hr. The total dose accumulated from a worker at 15' for 8 hours actually decreases to 776 mrem from 1310 mrem. The contact dose rate is less than a 10% increase in the margin to the regulatory limit of 5 rem. The 15' dose has decreased due to the conservative omega (Ω) scaling that was applied to the original calculation's contact dose rate. The original methodology utilized an infinite (∞) source geometry multiplied by a function of the ratio of the square of the distance from the source. The Blockage of HSM Air Inlets and Outlets has had its dose consequence increase from a value of 400 mrem to 584 mrem. This is less than a 10% increase in the margin to the 5 rem limit. There have been no changes to the Dry Shielded Canister Leakage Analysis.

4. ☐ YES ☒ NO Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR/USAR?

Justification: CCNPP ISFSI Cask Loading and HSM operations have no non-accident analysis of malfunctions of SSCs important to safety (with dose consequences) evaluated in the USAR.

5. ☐ YES ☒ NO Create a possibility for an accident of a different type than any previously evaluated in the UFSAR/USAR?

Justification: This activity only calculates the dose rates (and doses) associated with the CCNPP ISFSI Cask Loading and HSM facility yielded by utilizing new calculational methodology and does not create a possibility for an accident of a different type than any previously evaluated in the USAR.

Proposed Activity NO: ES200101042	**50.59 Log No.:**	
	72.48 Log No.:	**SE00160**

6. ☐ YES ☒ NO

Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR/USAR?

Justification: This activity only calculates the dose rates (and doses) associated with the CCNPP ISFSI Cask Loading and HSM facility yielded by utilizing new calculational methodology and does not create the possibility for a malfunction of any SSC.

7. ☐ YES ☒ NO

Result in a design basis limit for a fission product barrier as described in the UFSAR/USAR being exceeded or altered?

Justification: This activity only calculates the dose rates (and doses) associated with the CCNPP ISFSI Cask Loading and HSM facility yielded by utilizing new calculational methodology and does not calculate or affect any design basis limit for a fission product barrier.

8. ☐ YES ☒ NO

Result in a departure from a method of evaluation described in the UFSAR/USAR used in establishing the design bases or in the safety analyses?

Justification: This product is a calculation of the dose rate (and doses) associated with the CCNPP ISFSI Cask Loading and HSM operations that utilize a new NRC-approved methodology (MCNP) that provides more precise results (See attached writeup) and hence is not considered a departure from a method of evaluation described in the USAR. Several NRC SER's have been issued for both shielding and criticality licensing analysis that utilize MCNP. One of these is USNRC, "Safety Evaluation Report Concerning the Private Fuel Storage Facility," Docket 72-22, 9/29/2000. Utilizing this NRC approved methodology (MCNP) provides more precise results and its use (alone) does not necessitate prior approval. A list of other MCNP SER's is included as Attachment 1.

Summary: (For NRC Report, provide a brief overview) CCNPP has performed new calculations that will supersede the original design basis dose calculations and models for the Independent Spent Fuel Storage Instillation (ISFSI) 24P Dry Shielded Canister (DSC), Transfer Cask (TC) and Horizontal Storage Module (HSM). These calculations use improved methodology (3D vs 1D etc.) and have been performed with the Los Alamos National Laboratory Monte Carlo N-Particle Transport Code System MCNP4C. Several NRC SER's have been issued for both shielding and criticality licensing analysis that utilize MCNP. The calculations include measured benchmarks, are conservative, and did not 'Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR'.

Attachment 1

List of MCNP SER's

1) Safety Evaluation By The Office Of Nuclear Reactor Regulation Related To Amendment No. 182 To Facility Operating License No. Dpr-28, Vermont Yankee Nuclear Power Corporation, Vermont Yankee Nuclear Power Station, Docket No. 50.271.

2) Transnuclear West Standardized Advanced Nuhoms ® System Horizontal Modular Storage System For Irradiated Nuclear Fuel Docket No. 72-1029 Model Nos. Standardized Advanced Nuhoms ® -24P Transnuclear West, Inc. Certificate Of Compliance No. 1029

3) Safety Evaluation Report For Disposal Criticality Analysis Methodology Topical Report, Revision 0 June 2000 U.S. Nuclear Regulatory Commission Office Of Nuclear Material Safety And Safeguards Washington, DC 20555-0001.

4) Safety Evaluation Report Docket No. 72-1026 Fuelsolutions Storage System Certificate Of Compliance No. 1026

5) Safety Evaluation By The Office Of Nuclear Reactor Regulation GE Nuclear Energy Topical Report NEDC-32983P "General Electric Methodology For Reactor Pressure Vessel Fast Neutron Flux Evaluations" Project No. 710

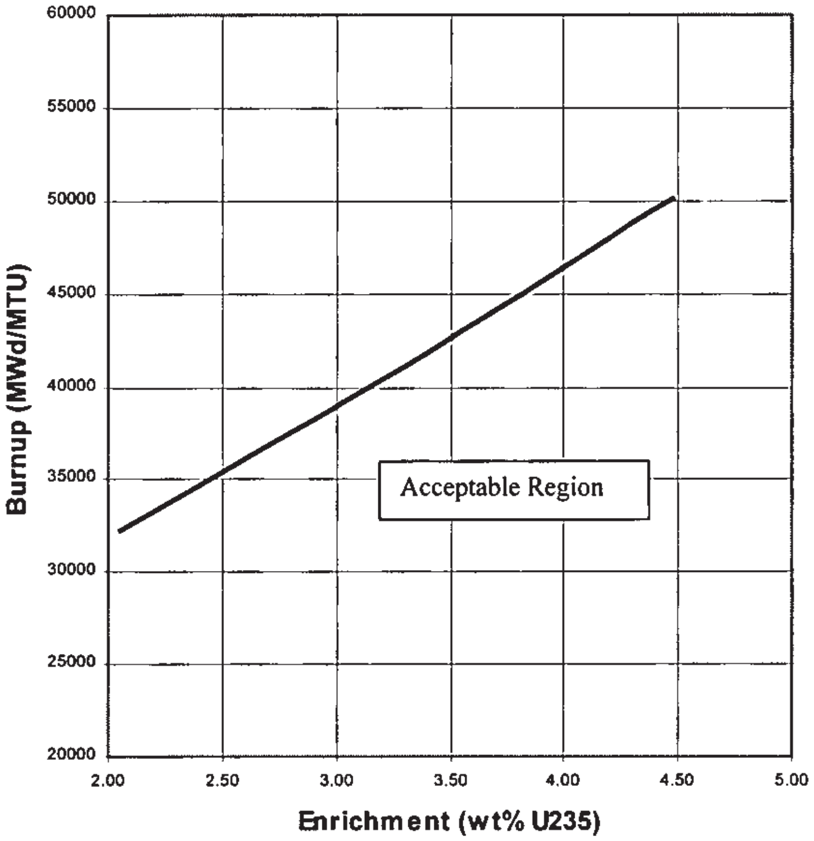
6) Holtec International Hi-Storm 100 Cask System Safety Evaluation Report

7) Safety Evaluation Report Docket No. 71-9302 Model No. Nuhoms ® -MP197 Transportation Package Certificate Of Compliance No. 9302 Revision No. 0

Attachment 2

ISFSI SAR Figure 7.2-1

Figure 7.2-1
24P Design Basis Radiological Limit Curve for 660 Watts/Assembly



*	This curve is valid for all assemblies that have cooled at least 9 years.
**	All assemblies loaded into DSC must meet the source spectra requirements of Technical Specification 2.1
***	This bounding radiological limit curve may be superceded with bundle specific source spectra calculations based on actual cooling times.

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Proposed Activity No: ES200100208-000 **50.59 Log No.:**

72.48 Log No.: SE00162

Does the Proposed Activity:

☐ YES ☒ NO Require a License Amendment for a change to the Technical Specifications/License Conditions?

Based on this 10 CFR 50.59/10 CFR 72.48 evaluation, does the Proposed Activity:

☐ YES ☒ NO Require a License Amendment because it meets one (or more) of the eight (8) criteria of 10 CFR 50.59(c)(2)/10 CFR 72.48(c)(2)?

Prepared by: Shane R. Gardner Department: TK-30-01 Date: 9/2/09

Printed Name and Signature

☐ YES ☒ NO Are cross-disciplinary concurrence reviews needed?

If "YES", document completion of these reviews below:

Responsible Individual	**Responsible Individual**	**Responsible Individual**
Print Name	Print Name	Print Name
Print Organization	Print Organization	Print Organization
/	/	/
Signature Date	Signature Date	Signature Date

INDEPENDENT REVIEWER: John R. Massari 9/2/09

Print/Signature Date

APPROVED: Phil I. Wengloski 9/2/09

Manager - Engineering Services or
Director - Nuclear Fuels Services

Print/Signature Date

ATTACHMENT 4, 10 CFR 50.59/10 CFR 72.48 EVALUATION FORM

Proposed Activity (Description):
The proposed activity is to authorize loading of undamaged, non-VAP fuel with stainless steel pins and/or vacancies into the 32P DSCs. The proposed activity includes adding administrative restrictions in the fuel selection procedure. The design change technical evaluation demonstrates the DSC will continue to meet its design requirements. Additional controls are introduced to limit the exposure and cooling time of the stainless pins such that the design basis shielding evaluation described in the USAR is unaffected. Also, controls are introduced that limit the configuration of vacancies such that the structural adequacy of the fuel assembly will be maintained during design basis accidents.

Reason for Activity:
As eligible fuel in the spent fuel pool is depleted there is a need to load reconfigured fuel assemblies. In the 2009 loading campaign Guardian fuel will be loaded (ECP-09-000054) and about one DSC worth of fuel containing inert stainless steel pins or vacancies. These assemblies are structurally intact and undamaged so they can logically be loaded into the ISFSI provided they meet all the design and licensing requirements. Other generically licensed NUHOMS systems allow storage of these kinds of assemblies, although fuel with vacancies is often stored as damaged fuel.

Affected UFSAR/USAR/ISFSI-USAR Described Design Functions:

- Confinement - The DSC design provides mechanical confinement of the stored fuel assemblies to prevent the dispersion of particulate or gaseous radionuclides from the fuel, and to maintain a barrier of helium around the fuel. The primary function of the DSC is to provide confinement of the spent nuclear fuel. This is achieved by the stainless steel shell and two inner cover plates (top and bottom ends) which are welded to the shell assembly. There are also outer cover plates (top and bottom) to further support the confinement integrity. The DSC confinement boundary is also designed to retain helium cover gas inside the DSC in order to prevent corrosion of the fuel cladding and formation of expansive oxides in the fuel itself during storage.

- Shielding - The DSC materials provide radiation shielding. The DSC provides shielding at its ends by the use of lead shield plugs. These plugs provide “As Low As Reasonably Achievable” (ALARA) dose rates at the top of the canister during drying and sealing operations and at the bottom for minimizing dose rates during DSC loading into the horizontal storage module (HSM) and at the HSM door during storage. The NUHOMS®-32P DSC has full-length egg crate plates and full-length steel rails with aluminum inserts. When compared to the NUHOMS®-24P DSC, the full-length egg crate plates provide additional shielding between the assemblies, and the rails provide additional shielding between the assemblies and the transfer cask. This additional shielding partly compensates for the additional spent fuel assemblies, so the dose rates outside the transfer cask are only slightly higher for the NUHOMS®-32P DSC than for the NUHOMS®-24P DSC. The dose rates outside the HSMs are lower for storage of a NUHOMS®-32P DSC than a NUHOMS®-24P DSC.

- Criticality Control - The DSC design provides for sub-criticality during the wet loading, DSC drying, and interim storage operations. This is accomplished by a combination of the physical properties of the fuel; fixed neutron absorbers in the NUHOMS®-32P DSC basket; 2,450 ppm soluble boron in the spent fuel pool water; and CCNPP administrative controls for fuel identification, verification, and handling. Optimum moderation was utilized for both internal and external moderation. The internal moderator was modeled with a soluble boron concentration of 2,450 ppm while the external moderator was modeled with pure water. The internal moderator is defined as that part of the moderator that is contained within the basket, while the external moderator is defined as that part of the moderator that is contained outside of the basket (between the basket to canister gap, between the canister to transfer cask gap, and the neutron shield).

- Fuel Support and Configuration Control (Fuel Retrievability) - The DSC internal basket assembly provides support for the spent fuel assemblies during normal operations. The DSC also provides configuration control related to post accident recovery of spent nuclear fuel. The DSC is designed so that the worst-case postulated accidents, including a cask drop, will not result in deformation of the Internal Basket Assembly or the DSC shell to such a degree that retrieval of intact fuel assemblies is not assured. The structural characteristics of the transfer cask (TC) and the DSC limit the deceleration loads on the fuel assemblies so that their integrity is assured in the worst-case drop accident. The guide sleeves establish storage compartments for 32 spent fuel assemblies within the DSC. The borated egg crate plates work together with boron dissolved in the fuel pool water (2,450 ppm) to maintain sub-criticality. The egg crate plates work together with the guide sleeves and rails to maintain the fuel assembly geometry. The egg crates and rails support the weight of the fuel assemblies when the DSC is in a horizontal position.

Effects on Design Functions

In order to determine the effects on the design functions of the proposed activity two new calculation were performed:

ATTACHMENT 4, 10 CFR 50.59/10 CFR 72.48 EVALUATION FORM

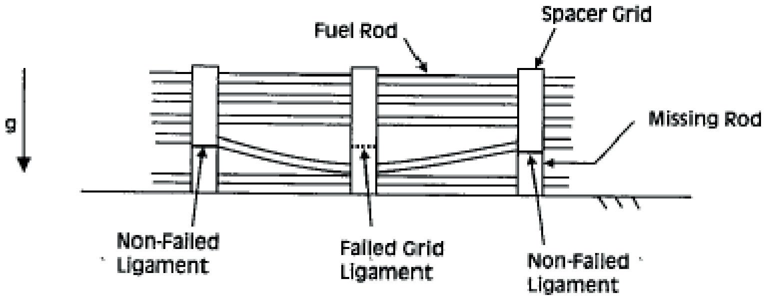
CA06534, “Accidental Drop Loading Evaluation of 14x14 Fuel Assembly with Missing Fuel Rods” (Hopper Elmore and Associates”

CA06367, “Comparison of the Radiological, Thermal, and Reactivity Characteristics of Assemblies with Missing or Inert Fuel Rods with the 32P ISFSI DSC Design Bases”

ES2001000208-000 and the associated 72.48 screen determined that none of the affected design functions were adversely affected. The following description summarizes the results of that evaluation and screen.

Calculation CA06534 analyzes the affect on cladding integrity for a fuel assembly containing missing fuel rods vacancies. The analysis is performed for the limiting design basis horizontal cask drop accident. The loads on the fuel assembly correspond to the design requirement of 75g. The analysis was performed similar to a previous calculation, for the 24P, CA05797, “DSC Horizontal Drop – Fuel Rod Cladding Integrity During Impact with a Broken Spacer Grid Fragment.” Essentially CA06534 is an extension of CA05797 to address vacancies.

Calculation CA06534 determined that the stress in horizontal spacer grid ligaments is excessive when vacancies are present and ligaments would fail. This is in addition to the failure of the vertical spacer grid ligaments previously determined in CA05797. The analysis determined that the likely scenario would result in failure of all horizontal spacer grid ligaments and the fuel rod would fall through the vacancy and strike the next fuel rod. However, to bound the horizontal spacer grid ligament failure CA06354 conservatively assumes that only one ligament fails resulting in the fuel rod being pinned by adjacent grid spacers and the rod deflecting until it strikes the fuel rod below. This scenario is depicted by the following figure extracted from CA06534:



The analysis determined that the maximum cladding stress excessive if more than two vacancies were present in any one column. The limiting configuration was determined to be up to two vacancies in one column and the maximum cladding stress was 68.5 ksi, which is less 77.5 ksi determined for the vertical ligament failure/grid collapse scenario in CA05797 and also less than the 0.9 x 80.5 ksi (USAR 8.2.5.2) stress allowable for irradiated zircaloy. As a result, the proposed activity, with the limitation of up to 2 vacancies in any on column, is bounded by the existing cladding stress analyses as described in USAR 8.2.5.2. It is noted that the analyses described in USAR 8.2.5.2 relate to the 24P and not the 32P.

Calculation CA06367 reanalyzes the criticality, shielding and decay heat effects of the proposed change. With respect to criticality, the analyses are performed using the bounding case from calculation CA06227, “Criticality Analysis of the NUHOMS 32P for Calvert Cliffs ISFSI”, which is the analysis of record. The computer model, code, and code revision are the same. The input to the computer model is modified to introduce inert stainless steel rods and vacancies. The material specification for stainless steel is based on the material reference in CA06227. The criticality analysis considers a number of variation in the configuration of the inert stainless steel rods and vacancies to identify the most reactive geometry. The calculated k-effective values are added to two KENO standard deviations resulting in the maximum k-effective with 95% probability and 95% confidence (95/95). The 95/95 k-effective values calculated all remain less than the previously determined bounding value of 0.9412. Because the replacement of fuel rods reduces the available

ATTACHMENT 4, 10 CFR 50.59/10 CFR 72.48 EVALUATION FORM

fissionable material the k-effective would be expected to decrease with the number inert stainless steel rods and vacancies. This was confirmed by calculation CA06367. The calculation also evaluates a case where stainless steel rods are concentrated on the periphery of the basket to maximize reflect and therefore reduce system leakage. Calculation CA06367 further evaluates the impact on the off-normal criticality calculation models in CA05896, “Criticality Analysis for Fuel Misloads and Accidents” and confirm the bounding k-effective of 0.9413 is not exceeded.

The effects on the shielding analysis due to the introduction of activated stainless steel rods were evaluated in CA06367 by performing a comparative analysis of the gamma source terms. The analysis is performed by using the design basis ORIGEN2 models from CA05803, “ISFSI 24P Assembly Insertion Requirements.” The computer model, code and code revision are the same. The input to the computer model is modified to introduce stainless steel rods. The material specification for stainless steel is based on the material reference in CA05803. Calculation CA06367 determines an energy dependent dose rate response function using the MCNP models from CA06292, “NUHOMS 32P Radiation Dose Rates for Loading and Transfer.” The response function provides a means to compare source terms on a dose rate basis. The response function allows consideration of the energy dependency of the source and the effects of shielding. The calculation analyzed several cases that varied the number of activated steel rods, exposure and cooling time, both for the steel rods and the host assembly, such that the resultant dose rates are bounded by the dose rate from the design basis assembly source. The results summary from CA06367 are presented below:

SS Pin Exposure MWd/MTU	# SS Pins Allowed by Years after 660W			
	0 years	1 year	2 year	3 year
20000	0	12	21	30
30000	0	7	13	18
40000	0	5	9	13

As the results indicate the configurations analyzed do not exceed the dose rate due to the design basis assembly source of 35.9 mrem/hr. Therefore the design basis gamma source in USAR 12.7.2.1 is bounding and hence the radiation shielding analyses in USAR 12.7.6 and 12.8.2.5.3 are unaffected by the proposed change.

From the ORIGEN2 calculations performed to analyze the gamma source terms, the decay heat due to the activated steel rods was also evaluated in CA06367. The decay heat produced by irradiated stainless steel is typically far less than the actinides and fission products present in spent fuel. This was confirmed by CA06367. The results show that decay heat produced by an activated steel rod quickly falls below the design basis heat of 660 watts/176 after about 4 years. The analysis of CA06367 also confirms that the thermal source of an activated stainless steel rod decays faster than a spent fuel rod, thus ensuring that the decay heat of the spent fuel rod will always be bounding.

As a result of the evaluations performed in CA06354 and CA06367 it is concluded that there are no adverse affects on the design functions of the 32P DSC. However, these conclusions are predicated on criteria restricting the configuration of the inert stainless steel rods and vacancies. To implement these criteria new procedures are required for fuel assembly qualification and these procedures are described in the USAR and constitute an adverse affect on a method of controlling design basis functions, thus requiring evaluation under 10 CFR 72.48.

ATTACHMENT 4, 10 CFR 50.59/10 CFR 72.48 EVALUATION FORM

UFSAR/USAR/ISFSI-USAR Sections reviewed where relevant information was found:	Tech Spec Sections reviewed where relevant information was found:
3.1.1, 3.1.1.3, Table 3.1-3, 3.3.1, Table 3.3-1, Table 3.3-3, 7.2.1, 8.2.1, 8.2.5.2, 9.4.1, 9.4.1.1, Table 9.4-1, 12.3.3.4.3, 12.3.3.4.4, 12.3.3.4.5, 12.3.6, Table 12.3-2, Table 12.3-3, 12.7.2.1, 12.7.6, 12.8.2.5, 12.8.2.5.3, 12.8.2.8	2.1, 2.4, 3.11

Does the proposed activity:

1. ☐ YES ☒ NO ☐ N/A Result in more than a minimal increase in frequency of occurrence of an accident previously evaluated in the UFSAR/USAR/ISFSI-USAR?

Justification:

The 32P DSC and the fuel assemblies to be stored are passive structures. The proposed activity has been evaluated to show that the fuel assembly can withstand the limiting and bounding cask drop accident. The 32P DSC and fuel assembly have no ability to initiate an accident or off-normal event. Therefore, the presence of inert stainless steel rods and/or vacancies cannot directly influence the expected frequency of an accident or off-normal event as defined in the USAR.

2. ☐ YES ☒ NO ☐ N/A Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component (SSC) important to safety previously evaluated in the UFSAR/USAR/ISFSI-USAR?

Justification:

The proposed activity has been evaluated to show the cladding can withstand the limiting and bounding cask drop accident. A malfunction of the cladding is defined by a loss of the structural integrity of the zircaloy due to excessive stress.

The maximum cladding wall stress calculated in CA06354 is 68.5 ksi for the limiting condition of 2 vacancies in any one column of the fuel array. That value is bounded by the previously determined value of 77.5 ksi in CA05797 due to a collapsed grid. Since the probability of cladding failure is proportional to the maximum cladding wall stress it is evident that the proposed activity does not result in an increase in the likelihood of occurrence of a malfunction of an SSC important to safety.

The proposed activity does not influence any other SSC malfunctions beyond cladding failure.

ATTACHMENT 4, 10 CFR 50.59/10 CFR 72.48 EVALUATION FORM

Does the proposed activity:

3. ☐ YES ☒ NO ☐ N/A Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR/USAR/ISFSI-USAR?

Justification:

As described above the proposed activity was evaluated against all the design requirements and safety analyses, including dose consequences associated with the cask drop and DSC leakage events. The analysis in CA06354 showed that cladding integrity is unaffected during a cask drop as long as no more than 2 vacancies are allowed in any one column of fuel rods. Also, CA06367 determined that the activation of stainless steel rods does not generate any radioactive gases, and actually reduces the activity available for release inside the DSC. Therefore, based on these evaluations it is concluded that there can be no increase in the dose consequences in the DSC leakage accident.

Calculation CA06367 also analyzed the impact of introducing activated stainless steel rods on the fixed radiation source term. The activated stainless steel rods will reduce the neutron source by replacing actinides and fission products in the spent fuel rods with the activated light elements which don't produce neutrons. Thus, the neutron dose consequences due to the proposed activity will decrease. The calculation in CA06367 also analyzed the gamma source of the activated stainless steel trough consideration of the affect on dose rates on the transfer cask. The analysis determined the limiting number of activated stainless steel pins, exposure and cooling time, such that the dose consequences associated with the transfer cask will not exceed the current design and licensing bases. Therefore, by design there can be no increase in the dose consequences in the DSC cask drop accident.

In general, because of the evaluations described above the design of the proposed activity will prevent an adverse affect on all the dose consequences for the 32P system. The limitations necessary to maintain the design bases will be implemented during the selection and qualification of fuel assemblies containing stainless steel rods and/or vacancies.

4. ☐ YES ☒ NO ☐ N/A Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR/USAR/ISFSI-USAR?

Justification:

As described above the proposed activity will not increase the source terms and by design will ensure the dose consequences are bounded or less than the current design bases. As a result, there is no increase in any dose consequences for the 32P system.

5. ☐ YES ☒ NO ☐ N/A Create a possibility for an accident of a different type than any previously evaluated in the UFSAR/USAR/ISFSI-USAR?

Justification:

Since the proposed activity involves a fuel assembly with inert stainless steel rods and/or vacancies and which is a passive structure that is compliant with the structural design criteria specified in the USAR, there is no possibility for the inert stainless steel rods and/or vacancies to initiate an accident. The fuel assembly must, by Technical Specification, be intact and undamaged. The proposed activity has been evaluated on this basis and has shown the fuel assembly and 32P DSC will continue to meet the design requirements. Because the fuel assembly is intact and undamaged the loading and unloading operations are unaltered. The proposed activity does not create a possibility of an accident of a different type than previously evaluated in the USAR.

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6. ☐ YES ☒ NO ☐ N/A Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR/USAR/ISFSI-USAR?

Justification:

The proposed activity could lead to a malfunction of the cladding due to a different kind of failure mode. This occurs during the limiting design basis cask drop event where calculation CA06354 has shown that vertical grid spacer ligaments could fail due to excessive stress. This new mode is not a malfunction with a different result because the result or effect is the same as and bounded by that previously described in the USAR. That is, the result is the same, provided the configuration of vacancies is limited to no more than two in any one column, to previous evaluations that have shown there will be no cladding failure. Also, the result remains bounded by the confinement evaluation described in USAR 12.8.2.8, which conservatively assumes 100% cladding failure. Thus any hypothetical failure, although shown to not occur, would be bounded by confinement evaluation previously described in the USAR.

7. ☐ YES ☒ NO ☐ N/A Result in a design basis limit for a fission product barrier as described in the UFSAR/USAR/ISFSI-USAR being exceeded or altered?

Justification:

The proposed activity relates to three design basis limits for fission product barrier as described in the USAR: k-effective < 0.95, cladding stress < 80.5 ksi and fuel assembly decay heat < 660 watts. In calculation CA06367 criticality control was reanalyzed to show that the resultant k-effective due to any reconfiguration of fuel assemblies with inert stainless steel rods and/or vacancies will always be less than current safety analyses as described in the USAR. To address the structural integrity of the fuel assembly calculation CA06354 determined the maximum cladding stress due to failure of a horizontal grid spacer ligament due to the limiting design basis cask drop accident. The calculated maximum cladding stress was determined to be bounded by existing analyses described in the USAR for the 24P for the collapsed grid. Thus, with respect to criticality control and fuel assembly structural integrity the proposed change does not exceed or alter the design basis limit for fission product barrier as described in the USAR.

The radiation source term evaluation for the proposed activity indicates that additional cooling time is required for fuel assemblies with irradiated stainless steel rods, which indirectly requires the assembly heat load to be less than 660 watts. The assembly heat load will also be reduced as the thermal source from the irradiated stainless steel rod will be less than the spent fuel rod it replaced. The reduction in the heat load limit is necessary not to protect the cladding fission product barrier as the design basis limit for a fission product barrier is intended (i.e. to maintain the peak cladding temperature limits), but rather to ensure the design basis gamma source is not exceeded. Therefore, the change in the assembly heat load does not result in a design basis limit for a fission product barrier being exceeded or altered.

8. ☐ YES ☐ NO ☒ N/A Result in a departure from a method of evaluation described in the UFSAR/USAR/ISFSI-USAR used in establishing the design bases or in the safety analyses?

Justification: See 10CFR72.48 screening form attached to ES200100208-000. The methods used to evaluate the proposed change are fully compliant with the methods of evaluation as described in the USAR sections 3.1.1.3, 7.2.1, 12.3.3.4.3, 12.3.3.4.4 and 12.3.3.4.5.

Summary: (For NRC Report, provide a brief overview)

The purpose of this activity is modifying the design of the NUHOMS-32P system to allow storage of irradiated and unirradiated inert stainless steel rods and/or missing fuel rods. To allow storage of such assemblies new design analyses have been performed to evaluate the effects on the design basis functions.

The presence of vacancies was analyzed to ensure the structural integrity of the fuel assembly and

ATTACHMENT 4, 10 CFR 50.59/10 CFR 72.48 EVALUATION FORM

cladding will be maintained during the limiting design basis horizontal cask drop accident. The evaluation perform determined the vacancy will lead to failure of the horizontal spacer grid ligaments, however the resultant cladding stress was determined to be bounded by the previous analyses and will remain within the stress allowable when no more than two vacancies are allowed in any one column of the fuel assembly. This limitation on vacancies will be controlled by modification of the fuel qualification and selection procedure. This configuration and result ensures the cladding fission product barrier is not challenged, is bounded by previous design analyses and therefore does not result in any undue risk to the health and safety of the public.

To verify the affect on criticality control, the design basis criticality analysis normal and off-normal models were reanalyzed by introducing inert stainless steel rods and vacancies. The new criticality model results demonstrated a decreasing trend in k-effective with the number of inert stainless steel rods and vacancies and all resulting k-effective values were found to be bounded by the existing safety analysis results. The criticality analyses considered the limitation of no more than two vacancies allowed in any one column and determined no further configuration limitations. Therefore, the proposed activity will have no adverse affect on the criticality control design function of the NUHOMS-32P system.

The proposed activity will introduce storage of activated stainless steel pins. The radiological source term of an assembly with activated stainless steel pins was compared to the design basis gamma assembly source term to ensure the dose rates for the NUHOMS-32P system will not be exceeded. The evaluation determined certain configuration requirements to meet this objective. The results of the evaluation show the dose rates will be maintained provided the number of activated stainless steel pins, their exposure and cooling time and additional host assembly cooling time is specified. These limitations on storage of assemblies with irradiated stainless steel pins will be controlled by modification of the fuel qualification and selection procedure. Also, thermal source of the activated stainless steel pin was analyzed and compared to the existing design basis thermal source. The results show the thermal source of an activated stainless steel pin is much less than the design basis spent fuel rod and will always be bounded by the design basis spent fuel rod for cooling times beyond four years.

In conclusion the proposed activity to store assemblies containing inert stainless steel rods and/or vacancies in the NUHOMS-32P system has been evaluated against all affected design requirements and functions. The evaluations have shown there are no adverse affects on the design bases and the existing design analyses will continue to bound the proposed activity. In support of this conclusion, several new limitations on the configuration of assemblies containing inert stainless steel rods and/or vacancies have been identified and will be incorporated into the procedures for the qualification and selection of fuel assemblies.

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(Page 1 of 3)

PROPOSED ACTIVITY NO: ES200100653, Supplement 000, Revision 0001

50.59 Log No.: N/A

72.48 Log No.: SE00163 Rev. 0001

USE OF NUHOMS-32P DRY SHIELDED CANISTER

Does the *Proposed Activity*:

☒ YES ☐ NO Require a License Amendment for a change to the Technical Specifications/License Conditions?

Based on this 10 CFR 50.59/10 CFR 72.48 evaluation, does the *Proposed Activity*:

☒ YES ☐ NO Require a License Amendment because it meets one (or more) of the eight (8) criteria of 10 CFR 50.59(c)(2)/10 CFR 72.48(c)(2)?

Prepared by: L. O. Wenger L. O. Wenger Department: 4A-11-07 Date: 10/25/05
PRINTED NAME AND SIGNATURE

☒ YES ☐ NO Are cross-disciplinary concurrence reviews needed?
If 'YES', document completion of these reviews below:

Responsible Individual	**Responsible Individual**	**Responsible Individual**
J. Kilpatrick	R. Beall	G. Gryczkowski
PRINT NAME	PRINT NAME	PRINT NAME
J.G. KILPATRICK	NFM	FOSU
PRINT ORGANIZATION	PRINT ORGANIZATION	PRINT ORGANIZATION *
10/27/05	10/27/05	10/28/05
SIGNATURE DATE	SIGNATURE DATE	SIGNATURE DATE

Responsible Individual	**Responsible Individual**	**Responsible Individual**
M. Massoud	J. Massari	G. Tesfaye
PRINT NAME	PRINT NAME	PRINT NAME
FOSU	FOSU	LICENSING
PRINT ORGANIZATION	PRINT ORGANIZATION	PRINT ORGANIZATION
Mahmoud Massoud 10/28/2005	10/28/05	10/28/05
SIGNATURE DATE	SIGNATURE DATE	SIGNATURE DATE

INDEPENDENT REVIEW: Print/Signature: H. D. Enoch / Harvey D Enoch 10/28/2005
INDEPENDENT REVIEWER Date

APPROVED: Print/Signature: J. J. McHale J McHale 10/28/05
Manager, CCNPP Engineering Services or Director, Nuclear Fuels Services Date

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 05- 032 Date: 10/28/05

Recommend Approval ☒ Recommend Disapproval ☐ POSRC CHAIRMAN Date

Approved ☒ Disapproved ☐ Joseph E. Pollock Date 10/28/05
PLANT GENERAL MANAGER

* J. Massari verified that comments provided by G. Gryczkowski via e-mail on 10/3/05 were incorporated.

PLANT GENERAL MANAGER

The NSRB Engineering Sub-Committee has reviewed this evaluation according to NS-2-100.

Full NSRB review required? ☐ YES ☒ NO

Print/Signature: ____________________ Date: 4/14/06

NSRB Engineering Sub-Committee

If '**YES**', NSRB Meeting No. ____________________

PROPOSED ACTIVITY NO: ES200100653, Supplement 000, Revision 0001

50.59 Log No.: N/A

72.48 Log No.: SE00163 Rev. 0001

USE OF NUHOMS-32P DRY SHIELDED CANISTER

Proposed Activity (Description): See Attachment 1

Reason for Activity: See Attachment 2

Affected UFSAR/USAR Described Design Functions: See Attachment 3

UFSAR/USAR Sections reviewed where relevant information was found:	Tech Spec Sections reviewed where relevant information was found:
(See Attachment 4)	All Sections of the Tech. Spec. were reviewed. Specifically: 2.1, 2.4, 3.1.1, 3.2.1.1, 3.3.1.1, 3.4.1.1, 4.2.1.1, 4.2.1.2, and 4.4.1.2.

Does the *proposed activity*:

1. ☐ YES ☒ NO ☐ N/A — Result in more than a minimal increase in frequency of occurrence of an accident previously evaluated in the UFSAR/USAR?

   Justification: See Attachment 5

2. ☐ YES ☒ NO ☐ N/A — Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component (SSC) important to safety previously evaluated in the UFSAR/USAR?

   Justification: See Attachment 6

10 CFR 50.59/10 CFR 72.48 EVALUATION FORM

(Page 3 of 3)

PROPOSED ACTIVITY NO: **ES200100653, Supplement 000, Revision 0001**

50.59 Log No.: **N/A**

72.48 Log No.: **SE00163 Rev. 0001**

USE OF NUHOMS-32P DRY SHIELDED CANISTER

Does the *proposed activity*:

3. ☐ YES ☒ NO ☐ N/A Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR/USAR?

Justification: See Attachment 7

4. ☐ YES ☒ NO ☐ N/A Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR/USAR?

Justification: See Attachment 8

5. ☐ YES ☒ NO ☐ N/A Create a possibility for an accident of a different type than any previously evaluated in the UFSAR/USAR?

Justification: See Attachment 9

6. ☐ YES ☒ NO ☐ N/A Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR/USAR?

Justification: See Attachment 10

7. ☒ YES ☐ NO ☐ N/A Result in a design basis limit for a fission product barrier as described in the UFSAR/USAR being exceeded or altered?

Justification: See Attachment 11

8. ☒ YES ☐ NO ☐ N/A Result in a departure from a method of evaluation described in the UFSAR/USAR used in establishing the design bases or in the safety analyses?

Justification: See Attachment 12

Summary: (For NRC Report, provide a brief overview) See Attachment 14

PROPOSED ACTIVITY DESCRIPTION

Activity:

The proposed activity is the use of a new design of Dry Shielded Canister (DSC), NUHOMS-32P, for storing spent fuel at the Calvert Cliffs Nuclear Power Plant's (CCNPP's) Independent Spent Fuel Storage Installation (ISFSI). The use of NUHOMS-32P DSCs will be made in addition to NUHOMS-24P DSCs. These two types of DSCs have the same external dimensions; however, the internal basket of the NUHOMS-32P DSC is designed to store 32 fuel assemblies. NUHOMS-32P Basket design is based on Transnuclear TN-68 design, which is approved by the NRC.

There are no physical changes being made to the Transfer Cask (TC) or the Horizontal Storage Modules (HSMs). (Note: The DSC Upender L-Bracket has been redesigned to support the additional weight of the NUHOMS 32P DSC. This activity will be issued under ES200100653-001-0000.) All the major steps for loading a DSC (vacuum drying, welding, etc.) are the same for the two DSC systems.

The activity also consists of the following additional fuel assembly changes, which have been included in the design analysis of NUHOMS-32P DSC.

- Increase of fuel assembly weight from 1,300 lbs. to 1,450 lbs. This accounts for the weight increase during irradiation and the possible storage of control components with the assembly.
- A limiting fuel assembly mass of 0.400 MTU, compared to the previously used nominal value of 0.386 MTU. This bounds all standard CE 14x14 fuel assemblies used at CCNPP.
- Increase in fuel assembly neutron source design basis in calculating the radiation dose, in order to account for higher burnups in lower enriched assemblies.

The changes being made to the ISFSI USAR consist of revising each existing section to add information about the NUHOMS-32P DSC where possible, and adding a new chapter, Chapter 12, which contains new information that is specific to the NUHOMS-32P DSC. The format of the new Chapter 12 follows the format of the CCNPP ISFSI USAR. It presents analyses to show that the NUHOMS-32P system meets all the requirements of 10 CFR 72.

Background:

The CCNPP ISFSI stores spent fuel assemblies inside a Dry Shielded Canister in a concrete reinforced horizontal storage module.. The spent fuel assemblies are first discharged into the spent fuel pool, where they are allowed to decay and cool. The assemblies are then transferred to and stored within the DSC which is loaded inside the Transfer Cask. The interior of the DSC is then vacuum dried, filled with the inert gas medium, helium, and sealed by welding. The DSC and Transfer Cask are then transported to the HSM.

The DSC is aligned with the storage location in a HSM and pushed in by a ram mechanism. The spent fuel decay heat is removed from the HSM by natural air circulation. The storage of the fuel assemblies is a completely passive system.

NUHOMS-32P DSC Design:

NUHOMS-32P DSC is designed to store 32 spent fuel assemblies, which is eight more assemblies than NUHOMS-24P DSC. The external and internal dimensions of the DSC shell are the same in both the designs. The NUHOMS-32P storage capacity is increased by reducing the size of and the space between the storage locations. Since the external dimensions of the DSCs are the same, the TC and HSM designs are not altered, however, their capability to accommodate the NUHOMS-32P is established through analyses.

Major components of the NUHOMS-32P DSC consist of guide sleeves, egg-crate plates (borated and un-borated), peripheral steel rails with aluminum inserts, end shield plugs, DSC shell, and end cover plate (see figures 1 and 2). Location of the vent and siphon ports have been moved from the DSC shell wall (in NUHOMS-24P DSC) to the DSC shield plug (in NUHOMS-32P DSC). This change is made to improve the welding operation of the shield plug to the DSC shell. Similar to the NUHOMS-24P DSC, the NUHOMS-32P DSC serves to provide confinement, shielding, criticality control, and structural integrity of the fuel during transit and storage of the spent fuel assemblies.

The NUHOMS-32P DSC's basket consists of 32 stainless steel guide sleeves (one for each spent fuel assembly) and uses stainless steel and aluminum (borated and unborated) plates, which are designed similar to an egg-crate, to support the guide sleeves. Both the guide sleeves and the egg-crate components run the full length of the DSC cavity. This allows the guide sleeves to be in contact with the egg-crate components over the entire length of the DSC cavity. There are peripheral steel rails around the egg-crate components. These steel rails maintain the position of the egg-crate components and the stainless steel guide sleeves within the DSC shell. As with the NUHOMS-24P DSC design, the basket assembly is not attached to the DSC shell walls or cover plates. Figures 1 and 2 illustrate the NUHOMS-32P DSC geometry and design.

Differences In Design:

The main difference between the NUHOMS-32P and NUHOMS-24P DSCs is the spent fuel storage capacity, namely the NUHOMS-32P can store 32 fuel assemblies compared to NUMHOMS-24P that can store 24 fuel assemblies. The NUHOMS-32P DSC is designed to accommodate the extra weight and thermal loads imposed by the larger number of fuel assemblies.

1. The NUHOMS-32P basket assembly is a different design than the NUHOMS-24P design. Specifically, the NUHOMS-32P DSC basket assembly design, differs from the NUHOMS-24P design as follows:

It has narrower guide sleeves,

- Center-to-center distance between guide sleeves is smaller,
- Neutron absorber material is provided between the guide sleeves to maintain the fuel configuration sub-critical,
- Additional radiation shielding is offered by the new basket assembly design, and
- The basket thermal resistance is lower and hence provides more efficient heat transfer.

2. The DSC shell for NUHOMS-32P has the same internal and external dimensions as the shell for NUHOMS-24P, but is slightly re-designed to support a future transportation license.
3. The DSC shell design accident pressure is increased to 100 psig.
4. The TC design remains the same as before, and has been analyzed for its ability to handle the NUHOMS-32P DSC.
5. The transfer vehicle (trailer, skid positioning system, and cask support skid) design remains the same as before, and has been analyzed for its ability to handle the NUHOMS-32P DSC.
6. The ram mechanism design remains the same as before, and has been analyzed to cope with the situation of a jammed NUHOMS-32P DSC.

7. The HSM design remains the same as before, and has been analyzed for its ability to handle the NUHOMS-32P DSC.
8. Increase of fuel assembly weight from 1,300 lbs. to 1,450 lbs (Note: the higher weight of 1450 lbs was incorporated by Technical Specification amendment for the 24P.) This accounts for the weight increase during irradiation and the possible storage of control components with the assembly. The larger weight of the fuel assembly has been used in the analyses.
9. A limiting assembly mass of 0.400 MTU, compared to the previously used nominal value of 0.386 MTU. This bounds all standard CE 14x14 fuel assemblies used at CCNPP. The larger mass has been used in the analyses.
10. A higher assembly neutron source has been used in calculating the radiation dose, in order to account for higher burnups in lower enriched assemblies.
11. The blocked vent accident response time at the HSM has been reduced from 24 hours to 12 hours. This reduced time is considered sufficient to unblock the vents.
12. The heavy haul road remains the same as before, and has been analyzed to handle the NUHOMS-32P DSC.
13. The support equipment for cask closing operation (vacuum drying, welding, etc.), remains the same as before, and has been analyzed to support closing the NUHOMS-32P DSC.

Numerical Values of the differences are tabulated in Table 1.

Table 1
Differences Between the Existing and New Designs

Parameter	NUHOMS-24P DSC	NUHOMS-32P DSC
No. of Fuel Assemblies	24	32
Guide Sleeve Dimension	8.7"x 8.7"	8.5"x 8.5"
Guide Sleeve Spacing	10.36"	9.125"
Lead Shielding Thickness - Top Shield Plug	4.375"	4.00"
Top Shield Plug Top Casing Plate	0.375"	0.75"
Spent Fuel Pool Min. Boron 10 Concentration	1800 ppm	2450 ppm
Borated Aluminum Plates – Boron Content	None	0.01 g/cm^2
Fuel Assembly Mass	0.386 MTU (Nominal)	0.400 MTU (Maximum)
Neutron Source	2.23E+08 n/s/assy	3.3E+08 n/s/assy
Fuel Assembly Weight	1300 lbs(1)	1450 lbs
Loaded Dry DSC Weight	65.0 kips	91.0 kips
Total Heat Source (0.66 kW/assy)	15.84 kW	21.12 kW
DSC Shell Design Pressure (Accident Condition)	50 psig	100 psig
Blocked Vent Restoration Time	24 hours	12 hours
Optimum Moderation Assumption	Pure Water	Borated Water

(1) A Technical Specification Amendment was issued for 24P for a maximum weight of 1450.

Figure 1

Basket Geometry – NUHOMS 32P DSC

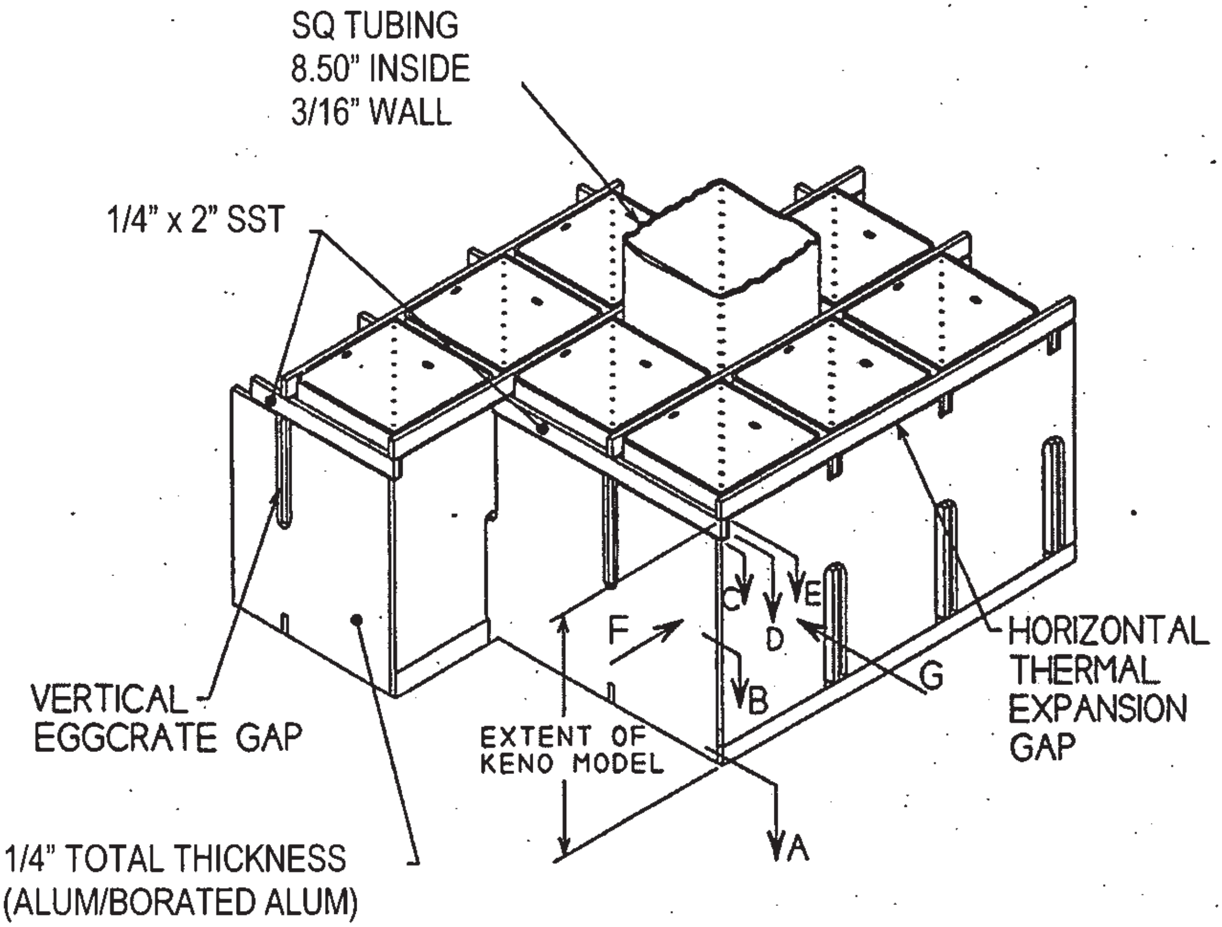
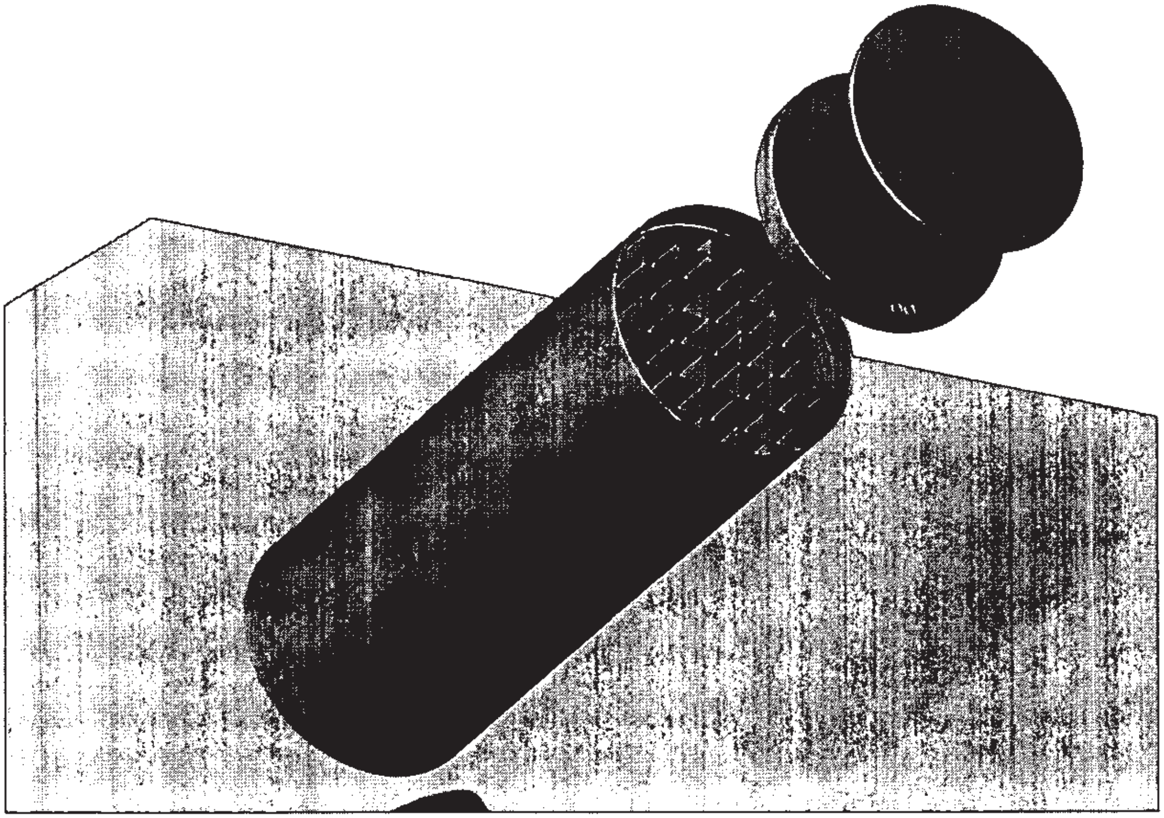


Figure 2

NUHOMS 32P DSC



Attachment 2
72.48 Log No. SE00163 – 0001 - USE OF NUHOMS-32P DRY SHIELDED CANISTER

REASON FOR ACTIVITY

This proposed activity will increase the dry storage capacity at the CCNPP ISFSI. A NUHOMS-32P DSC stores eight more spent fuel assemblies than a NUHOMS-24P DSC using the same external shell dimensions.

The spent fuel storage in ISFSI is temporary until the U.S. Department of Energy (DOE) begins to accept the fuel for permanent storage.

The ISFSI at the Calvert Cliffs Nuclear Power Plant (CCNPP) provides for the temporary dry storage of spent fuel assemblies within the DSCs. Each DSC is stored in one HSM. CCNPP has a license to build a total of 120 HSMs.

The NUHOMS-32P DSC design allows CCNPP to reduce the minimum number of canisters to be loaded each year from four (using the NUHOMS-24P DSC design) to three (with the NUHOMS-32P DSC design while increasing the total storage capacity of the ISFSI by 576 assemblies). This is expected to reduce the total annual radiological dose and loading costs and extend the total storage loading lifetime at the ISFSI by 6 years.

Attachment 3
72.48 Log No. SE00163 – 0001 - USE OF NUHOMS-32P DRY SHIELDED CANISTER

AFFECTED UFSAR/USAR DESCRIBED DESIGN FUNCTIONS

The NUHOMS-32P DSC stores eight more spent fuel assemblies than the previously analyzed NUHOMS-24P DSC. As a result, the following DSC design parameters are affected:

- Spacing between the fuel assemblies is decreased,
- Radiation sources are increased,
- Thermal sources are increased, and
- Overall weight is increased,

The effects on SSC design functions of utilizing the NUHOMS-32P DSC are summarized in Table 2. As seen in the table, the design functions of many SSCs and major components are adversely affected due to the changes identified above.

NUHOMS-32P DSC features that are designed to support its design functions are described below. A discussion of evaluation of adverse effects of changes in the above four parameters on the design functions is also presented in Table 2.

Attachment 3
72.48 Log No. SE00163 – 0001 - USE OF NUHOMS-32P DRY SHIELDED CANISTER

Table 2
Affected USAR SSC Design Functions

SSC and other Major Components	Design Function (USAR Table 3.3-1)	Affected By	Is the Effect Adverse?
Dry Shielded Canister (DSC) Internal Basket Assembly and Outside Shell	Criticality Control	Fuel Assembly Spacing Decrease	Yes
	Fuel Support (Structural Integrity)	Weight Increase	Yes
	Cover Gas Containment	Internal pressure affected by thermal source increase and reduction in free volume due to 8 more assemblies	Yes
	Radioactive Material Confinement	Cladding Temp. and internal pressure, affected by thermal source increase.	Yes
	Shielding (Surface Dose)	Radiation Sources Increase	Yes
Transfer Cask	On-Site Fuel Transport (Structural Integrity)	Weight Increase	Yes
	Shielding (Surface Dose)	Radiation Sources Increase	Yes
Horizontal Storage Module (HSM)	Shielding (Surface Dose)	Radiation Sources Increase	Yes
	DSC Support	Weight Increase	Yes
	DSC Tornado Missile Protection	None	No
	DSC Cooling	Thermal Source Increase	Yes
Ram Equipment	Ram force to cope with a DSC jammed in the HSM	Weight Increase	Yes
Transfer Equipment (Cask Support Skid and Positioning System Transfer Trailer Optical Alignment System Tractor)	Transfer Cask Movement, DSC Transfers	Weight Increase	Yes

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1. DSC Design Features That Support the Design Functions

The DSC consists of the outer canister shell and the internal basket assembly, and is classified in Section 3.2 of the CCNPP ISFSI USAR as important-to-safety per 10 CFR 72. The DSC provides containment (confinement), shielding, criticality control, configuration control related to fuel retrievability, structural support, and thermal safety functions during loading operations, transfer operations, and storage. It is designed to remain intact under all accident conditions identified in the CCNPP ISFSI USAR with no loss of function. Specific design functions of the DSC include the following:

1. Confinement - The DSC design provides mechanical confinement of the stored fuel assemblies to prevent the dispersion of particulate or gaseous radionuclides from the fuel, and to maintain a barrier of helium around the fuel. The primary function of the DSC is to provide confinement of the spent nuclear fuel. This is achieved by the stainless steel shell and two inner shield plugs (top and bottom ends) which are welded to the shell assembly. There are also outer cover plates (top and bottom) to further support the confinement integrity. The DSC confinement boundary also is designed to retain a helium cover gas inside the DSC in order to prevent corrosion of the fuel cladding and formation of expansive oxides in the fuel itself during storage.
2. Shielding - The DSC materials provide radiation shielding. The DSC provides shielding at its ends by the use of lead shield plugs. These plugs provide ALARA dose rates at the top of the canister during drying and sealing operations and at the bottom for minimizing dose rates during DSC loading into the horizontal storage module (HSM) and at the HSM door during storage.

   The NUHOMS-32P DSC has full-length egg crate plates and full-length steel rails with aluminum inserts. When compared to the NUHOMS-24P DSC, the full-length egg crate plates provide additional shielding between the assemblies, and the rails provide additional shielding between the assemblies and the transfer cask. This additional shielding partly compensates for the additional spent fuel assemblies, so the dose rates outside the transfer cask are only slightly higher for the NUHOMS-32P DSC than for the NUHOMS-24P DSC. The dose rates outside the HSMs are generally lower (except on the HSM Door) for storage of a NUHOMS-32P DSC than a NUHOMS-24P DSC.
3. Criticality Control - The DSC design provides for sub-criticality during the wet loading, DSC drying, and interim storage operations. This is accomplished by a combination of the physical properties of the fuel; fixed neutron absorbers in the NUHOMS-32P DSC basket; 2,450 ppm soluble boron in the spent fuel pool water; and CCNPP administrative controls for fuel identification, verification, and handling.

   Optimum moderation was utilized for both internal and external moderation. The internal moderator was modeled with a soluble boron concentration of 2450 ppm while the external moderator was modeled with pure water. The internal moderator is defined as that part of the moderator that is contained within the basket, while the external moderator is defined as that part of the moderator that is contained outside of the basket (between the basket to canister gap, between the canister to transfer cask gap, and the neutron shield).

4. Fuel Support and Configuration Control (Fuel Retrievability) - The DSC internal basket assembly provides support for the spent fuel assemblies during normal operations. The DSC also provides configuration control related to post accident recovery of spent nuclear fuel. The DSC is designed so that the worst-case postulated accidents, including a cask drop, will not result in deformation of the Internal Basket Assembly or the DSC shell to such a degree that retrieval of fuel assemblies is not assured. The structural characteristics of the transfer cask (TC) and the DSC limit the deceleration loads on the fuel assemblies so that their integrity is assured in the worst-case drop accident.

   The guide sleeves establish storage compartments for 32 spent fuel assemblies within the DSC. The borated egg crate plates work together with boron dissolved in the spent fuel pool water (2,450 ppm) to maintain subcriticality.

5. Heat Removal - During transfer operations, decay heat is removed by thermal radiation and gaseous conduction. During storage in the HSM heat is removed via natural circulation airflow. Decay heat is also radiated from the DSC surface to the heat shield and HSM walls where natural convection air flow and conduction through the walls removes the heat. The DSC also maintains the helium cover gas, which is required for corrosion control. This cover gas improves the thermal performance of the DSC.

2. <u>Effect of Decreased Fuel Assembly Spacing - Criticality Control Within DSC</u>

The criticality control in the NUHOMS-32P DSC, with the reduced fuel assembly spacing, is achieved by:

- Increasing the boron concentration in fuel pool water from 1,800 ppm to 2,450 ppm, and
- Providing poison plates between the fuel assembly storage locations within the DSC. The poison plates consist of borated aluminum plates with a minimum boron (B^{10}) areal density of 0.010 g/cm^2.

BORATED ALUMINUM INSPECTIONS

The effectiveness of borated aluminum is established through neutronic testing (Reference 14). Boron-10, which is enriched to 95 w/o, appears as discrete particles of AlB_2 uniformly distributed in the aluminum matrix. Test coupons are cut from the borated sheets and tested for neutron transmission. The transmission through the coupons is compared with transmission through calibrated standards. The effective boron-10 content of each coupon must be greater than 0.010 g/cm^2; otherwise the associated sheet is rejected.

CRITICALITY ANALYSES

Design basis criticality analyses were performed for Combustion Engineering (CE) design 14x14 non-Value Added Pellet (VAP) fuel assemblies containing UO_2 enriched up to 4.5 wt% U-235. The Upper Subcriticality Limit (USL), as defined in NUREG/CR-6361, Section 4, is calculated for the CCNPP fuel and NUHOMS-32P design through a series of calculations to be 0.9422. An effective multiplication factor (k_{eff}) that is less than USL ensures that the k_{eff} will be lower than the regulatory limit of 0.95 even when the biases and uncertainties in design parameters are taken into account.

The criticality within the NUHOMS-32P DSC has been analyzed for the following conditions:

- Normal operations
  - wet loading (optimum moderation)
  - dry storage (via moderator exclusion)

- Off-normal or accident conditions
  - fuel mis-load. Fuel mis-load analyses were performed for CE design 14x14 Value Added Pellet (VAP) fuel assemblies containing UO_2 enriched up to 5.0 wt% U-235, to determine how many VAP assemblies can be loaded without exceeding the k_{eff} limit.
  - off-normal poison plate thickness and accidental cask drop.

The criticality analyses are documented in Reference Series 1. The fuel mis-load analysis demonstrates that up to two fresh VAP assemblies can be loaded at optimum moderation (2,450 ppm boron concentration in the DSC) without exceeding the k_{eff} limit. These analyses demonstrate that the k_{eff} in all cases is below the USL of 0.9422. The analyses confirm that the Calvert Cliffs site-specific NUHOMS-32P design satisfies the requirements of 10 CFR 72.124 for normal, off-normal, and hypothetical accident conditions.

3. Effect of Increased Radiation Sources

The increase in radiation sources affects the radiation dose around the DSC, TC and HSM. Radiation sources inside a fully fuel-loaded NUHOMS-32P DSC are higher than those in a fully fuel-loaded NUHOMS-24P DSC because of the larger number of fuel assemblies. The gamma source per assembly is the same as before. The neutron source has been revised to a higher value to account for higher burnups in lower enriched assemblies. The change to an egg-crate basket design has increased the shielding provided by the DSC by the following:

- Addition of full length aluminum plates (borated and non-borated) between the guide sleeves and,
- Addition of full length stainless steel rails and aluminum rail inserts between the DSC stainless steel cylindrical shell and the outside guide sleeves.

The radiation dose analyses that support the NUHOMS-32P DSC upgrade addresses normal, off-normal and hypothetical accident conditions. They are essentially the same (see Attachment 12) in form and methodology to the design basis analyses that support the NUHOMS-24P DSC. Radiation dose rate values are calculated at various locations around the DSC, around the TC and around the HSM. The calculations are documented in Reference Series 4. The calculation results are tabulated in Table 3. They show that the gamma dose rates from the NUHOMS-32P DSC are similar to those from the NUHOMS-24P DSC. The neutron dose rates are higher due to the increased neutron source term. Outside the HSM, however, the neutrons do not dominate the total dose because the concrete in the HSM is an effective neutron shield. As such, the total doses outside the HSM are comparable for the NUHOMS-32P and NUHOMS-24P DSCs.

The total dose rate values around the HSM for NUHOMS-32P DSC are comparable to those of NUHOMS-24P DSC. Thus, the total storage time dose to the public remains unchanged. The ISFSI USAR (p. 7.4-2) also provides a calculation of the maximum dose outside the ISFSI site fence during seismic restraint installation. This dose is composed of two components: dose from all closed/loaded HSMs and dose from the HSM where the seismic restraint is being installed. CCNPP Drawing 84075 shows that the outer fence is 94 feet from the HSMs in the N/S direction, and 53 feet from the HSMs in the E/W direction. From the results of CA06327 and CA06058 the maximum doses at these locations from a site consisting of 120 loaded HSMs of 32Ps or 24Ps are 0.4 mrem/hr in the N/S direction, and 0.6 mrem/hr in the E/W direction. CA06327 indicates that the dose rate at a distance of 66.7 feet directly in front of a loaded 32P HSM with the door open 2 feet for seismic restraint installation is 17.1 mrem/hr. Note that this tally location is conservative because a person outside the fence in the N/S direction cannot get closer than 94' from the front of the HSM, and a person on the E/W side will not be in a direct line of sight of the open HSM. The ISFSI USAR indicates that seismic restraint installation requires a maximum of 5 minutes (0.08 hours) and can nominally be done in 1 minute (0.017 hours). Thus, a person just outside the fence would conservatively receive a maximum of 1.37 mrem (0.08 hr x 17.1 mrem/hr) during the seismic restraint installation process. If that person remained outside the fence for an entire hour during the seismic restraint installation, their dose would conservatively not exceed 1.97 mrem (1.37 mrem from seismic restraint installation + 0.6 mrem from the fully loaded ISFSI site). This is within the 10CFR20.105 requirement that no individual (member of the public) may receive greater than 2 mrem in any 1 hour. Therefore the dose to the public is not affected by the change to the NUHOMS-32P DSC.

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TABLE 3

DOSE ANALYSIS RESULTS - NOMINAL DOSE RATES (MREM/HR)

LOCATION	NEUTRON		GAMMA (PRI + SEC)			TOTAL	
	24P[c]	32P	24P[c]	32P		24P[c]	32P
DSC in HSM[d]							
1. HSM Wall or Roof	0.5	0.6	12	9.3		12.5	9.9
2. HSM Air Outlet	1	1.2	81	54.3		82	55.5
3. Center of Door	5	5.8	5	6.3		10	12.1
4. Doorway (Max., 1 ft. into opening)	621	1075	2619	2943		3240	4018
5. Air Inlet Vent	1	1	72	67		73	68
6. 1m from HSM Door	2	3	4	4.8		6	7.8
DSC in Transfer Cask							
1. Centerline[a] DSC Shield Plug (Flooded DSC)[e]	4	1.5	76	90.9		80	92
2. DSC Cover Plate (Dry DSC)[e]							
2.1 Center	45	48	96	98		141	147
2.2A Edge[b] (Wet Gap)	80	98	62	61		142	159
2.2B Edge[b] (Dry Gap)	124	134	136	137		260	271
3. Transfer Cask[f]							
3.1 Side	69	98	72	48		141	146
3.2 Top	6	8.5	1.0	1.4		7	10
3.3 Bottom	56	104	63	74		119	178
TC Drop Accident (@ 15')	80.6	117.9	15.8	9.7		97	127.6

(a) The DSC/cask annular gap is filled with water. All but the top 6 inches of the DSC inner cavity is filled with water.

(b) Nominal at edge of cover plate. The total dose rate is approximately a factor of 3 lower at the top edge of the transfer cask, and several times higher inside the dry annulus.

(c) NUHOMS-24P DSC does rates are from Calculations CA05924 and CA05925 (see Reference Series 4)

(d) NUHOMS-32P DSC in the HSM dose rates are from Calculation CA06293 (see Reference Series 4)

(e) NUHOMS-32P DSC dose rates are from Calculation CA06327 (see Reference Series 4)

(f) NUHOMS-32P DSC dose rates are from Calculation CA06292 (see Reference Series 4)

4. Effect of Increased Thermal Sources

The increase in thermal sources affects the pressure inside the DSC, temperatures in and around the DSC, and temperatures of the HSM concrete. The heat source in the NUHOMS-32P DSC is larger than the one in NUHOMS-24P DSC because of the increased number of fuel assemblies stored. The decay heat criterion used to load a fuel assembly into the DSC remains the same as before, namely 660 watts. With eight more assemblies the NUHOMS-32P DSC will have a heat source of 21.12 kW. The following design features have been included in the NUHOMS-32P DSC to improve the heat transfer and accommodate the larger heat source.

- Addition of full length aluminum plates (borated and non-borated) between the guide sleeves,
- Addition of stainless steel rails and aluminum rail inserts between the DSC stainless steel cylindrical shell and the outside guide sleeves.

Temperature and pressure analyses for the NUHOMS-32P DSC design has been performed as follows:

- Vacuum Drying
- Operating Pressures in the DSC
- Thermal Expansion of the DSC Components
- Thermal Analysis of the DSC in the Transfer Cask
- Thermal Analysis of the DSC in the HSM
- Thermal Analysis of the HSM
- HSM Exit Air Temperature

Analyses have been performed for the normal, off-normal and accident conditions, which are as follows:

- Cask drop during transit, which leads to the cladding rupture and escape of fuel gases into the DSC
- Blocking of all air vents in HSM for a period of 36 hours. The NUHOMS-24P design assumed that the HSM vents were blocked for 48 hours. A smaller time of blocking of the vents is used for NUHOMS-32P. This is justifiable because the vents are inspected every 24 hours. In the very low probability event where all three vents are found blocked cooling can be restored within 12 hours.

Analyses have been performed to determine the temperatures of the fuel assembly cladding within the DSC and temperatures around the DSC, TC and the HSM. The analyses are documented in the Reference Series 3, 5 and 6 respectively. Results of the analyses are tabulated in Table 4. The analyses confirm that the component temperature limits for the TC, DSC, and HSM are not exceeded for normal, off-normal, and accident conditions for the NUHOMS-32P DSC design at CCNPP. In addition, the HSM vents are required to be inspected at 24 hour intervals to maintain the vents unblocked to assure that the design temperature of the HSM concrete is not exceeded (see Technical Specifications Surveillance Requirement 4.4.1.2).

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Table 4
Thermal Analysis Results(j) – NUHOMS-32P DSC

Operating Condition	Cladding Temp (°F)	Helium Temp (°F)	DSC Pressure (psig)(i)	DSC Surface Temp (°F)	HSM Concrete Temp (°F)(c)	HSM Δ Air Temp (°F)	TC Lead Shield Temp (°F)	TC Neutron Shield Resin Temp (°F)
Vacuum Drying	750	-	3 Torr	-	-	-	-	-
Therm/Press Limit	1058	-	3 Torr	-	-	-	-	-
Transfer at 103°F	742	621	13.7 (88.5 (a))	-	-	-	370	277(g)
Therm/Press Limit	1058	-	100	-	-	-	620	280(h)
Transfer at - 3°F	664	(f)	-	-	-	-	265	224
Therm/Press Limit	1058	-	-	-	-	-	620	280(h)
Storage at 70°F	597	484	10.1	292	157	60	-	-
Therm/Press Limit	635	-	15	-	200	60	-	-
Storage at 103°F	620	509	10.8	323	201	64	-	-
Therm/Press Limit	1058	-	50	-	350	60(d)	-	-
Storage at -3°F	545	(f)	-	223	65	49	-	-
Therm/Press Limit	1058	-	-	-	350	60	-	-
Accident(b) at 103°F	<838(e)	725	98.5	571	387	-	-	-
Therm/Press Limit	1058	-	100	-	395	-	-	-

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a Transfer with accidental drop concurrent with failure of all fuel clad and release of gases.

b Assumes failure of all fuel clad and HSM vents blocked for 36 hours.

c Concrete temperatures are calculated at the roof, floor, and wall locations of the HSM. Of these the maximum value is listed for ambient temperatures of 103 °F and 70 °F. The maximum concrete temperature for the -3 °F case is not reported as it is bound by the 103 °F and normal storage cases. The minimum concrete temperature that the HSM could hypothetically experience is -3 °F.

d CA06299 (Exit Air Temp. and Bulk Air Temp. w/in the HSM) provides input into CA06302 (HSM Thermal Analysis – Max. Summer Temp.). CA06302 provides input to CA06306 (DSC Thermal analysis – Off Normal Storage Condition – Max. Summer Temp.). The results of CA06306 confirm that Fuel Cladding temperature remains well below thermal limits. ISFSI TS 3.4.1.1 establishes a maximum air temperature rise limit of ≤ 60°F. The basis for this TS Limit was selected to limit the hottest rod in the DSC to below 635°F at 70°F ambient air temperature. CA06299 predicts a maximum bulk air temperature rise across the DSC of 64°F. This predicted temperature rise should not be confused with the TS limit of ≤ 60°F. CA06306 establishes that Fuel Cladding temperature remains well below thermal limits. The TS Limit of ≤ 60°F is an operator guideline that establishes a TS Limiting Condition for Operation that initiates operator action to increase monitoring and/or corrective action for the on-set of an adverse condition.

e Reference Series 6, CA06304 only evaluates the blocked vent accident at 24 and 48 hour end points. The 48 hour case is conservatively reported for maximum fuel clad temperature.

f The off-normal minimum winter temperature cases (transfer and storage) are bounded by the results of the off-normal summer maximum temperature cases. No values are reported for the minimum winter cases. There is no temperature limit on the DSC Helium fill gas. Helium temperature is an input to determining the maximum internal pressure of the DSC under various operating conditions.

g Mass averaged bulk temperature of the neutron absorbing resin.

h Resin thermal limit was qualified by test.

i. The value of 15 psig is used as a design pressure for normal conditions; 30 psig is conservatively used in the associated stress calculations for normal conditions; 50 psig is used for off-normal conditions, and 100 psig is used for the associated stress calculations under accident conditions.

j. A bounding analysis was performed to determine the effect of increasing the mesh density from 5x5 and 6x6 to 14x14 in the fuel region. The results of the analysis are presented in the response to Question 8. The analysis shows that by increasing the mesh density to 14x14, the fuel cladding temperature increases by 1° F for normal conditions of storage and for the blocked vent accident condition. It is expected that the fuel cladding temperature will similarly increase for the other conditions listed in Table 4. These other conditions have significant margin to the thermal limits and are not the bounding cases. The limiting condition for the DSC pressure is the blocked vent condition. The gas temperature is conservatively assumed to increase 10°F which is greater than the temperature increase for any of the basket components. At this temperature, the DSC pressure would increase to 99.4 psig, which is below the design pressure of 100 psig.

5. Effect of Increased Weight - Structural Integrity

The structural integrity of the DSC, Plant Structures, TC, Access Route, and HSM is impacted by the increase in weight (See Table 5) of the DSC and the increase in temperatures (See Table 4). The weights of TC, Cask Support Skid and HSM remain unchanged.

The impact of these changes is analyzed on the structural design of the following components:

- DSC Basket Assembly
- DSC Shell
- TC
- Transfer Skid
- Lifting Devices, such as Yoke and trunnions (the plant facilities' evaluation of cranes and floor slabs is evaluated under a separate 50.59 review)
- Access Road Hydraulic Ram (for DSC insertion and removal), and
- HSM

The analyses are done for normal, off-normal and accident conditions. The analyses are documented in Reference Series 2 and 7. The bounding or the controlling calculated stress values, which correspond to the applicable accident cases, are tabulated in Table 6. The stainless steel thickness in the top shield plug top casing plate welded to the DSC shell is doubled from 0.375" thick to 0.75". To compensate for the increased thickness in the stainless steel, the lead thickness was reduced from 4.375" to 4.000" to maintain the over all thickness at 6.25". The results show that the stresses imposed by the heavier and hotter NUHOMS-32P DSC on all of these components are maintained within the allowable structural limits. Other variations between the NUHOMS-32P and NUHOMS-24P DSCs (such as lifting lugs and relocation of the vent and siphon ports to improve the welding operation of the shield plug to the DSC shell) have no impact on the structural integrity of the canister.

Hydraulic Ram System

The Hydraulic Ram System (HRS) is used to insert and extract the DSC into or from the HSM. The HRS has a maximum capacity of 80 kips.

The HRS design is not changed as a result of this proposed activity. The weight of the 32P DSC exceeds the weight of the 24P DSC.

A reduction of the ability to apply a ram force equal to the weight of the loaded 32P DSC, has the potential to adversely affect CCNPP's ability to extract a jammed DSC. However, the following features are incorporated into ISFSI procedures to minimize the potential for a jammed DSC:

- To help ease the sliding of the DSC in and out of the HSM, a dry film lubricant is applied to the support rails inside the HSM and the TC, both of which are in contact with the DSC during horizontal DSC transfer. The lubricant used is Perma-Slik RGAC, which the manufacturer certifies as providing a coefficient of friction of 0.02 to 0.04. Thus, the normal force required to insert or extract the NUHOMS-32P DSC is calculated (Reference Series 7, DSC Loading and Unloading Loads) using a nominal coefficient of friction (which credits the use of a dry film lubricant) of 0.04, to be 4.5 kips. This is well within the HRS capacity.
- The ISFSI Operating Procedure (Reference 15) limits the DSC insertion force to about 20 kips, which is approximately 25% of the NUHOMS-24P DSC weight and 21% of the 32P DSC weight. When a force larger than the setpoint is required, the DSC is considered jammed. Administrative controls are used at that point to remove, inspect and re-insert the DSC, or apply a larger force, i.e., up to 80,000 lbs. The administrative controls will ensure that the force applied for insertion will not exceed the maximum force that can be applied by the HRS to extract a jammed DSC.

It should be noted that the NRC has accepted (Reference 16) a HRS capacity of 80 kips for Generic License NUHOMS 32PT series DSCs, whose dry weights range from 88 kips to 101.2 kips. NUHOMS-32P DSC has a dry weight of 91 kips (95 kips is conservatively used in the Loading and Unloading Loads calculation). This weight is within the range of 32PT series weights previously approved by the NRC.

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Table 5
Weights and Loads – NUHOMS-24P vs. NUHOMS-32P

	Weight/Force (kips)	
	NUHOMS-24P	**NUHOMS-32P**
Fuel Assembly	1.3(a)	1.45
DSC - Empty	34.6	44.6
DSC - Loaded with Fuel(c)	65.0	91.0
Access Road(c) (Designed for 140 ton)	120.5 ton	131.3 ton

(a) A Technical Specification Amendment was issued for 24P for a maximum weight of 1450 lbs.

(b) The (conservative) transfer trailer load associated with moving a fuel-loaded NUHOMS-32P DSC is 132.35 ton.

(c) The actual loaded weight of 32P is 91.0 kips, 95 kips is conservatively used for structural analysis.

TABLE 6

Maximum Calculated Stresses for 32P DSC

DSC SHELL ASSEMBLY	TYPE OF STRESS(*)	MAXIMUM CALCULATED STRESSES FOR LEVEL D (ksi)	LEVEL D ALLOWABLE STRESS (ksi)
		32P	
Shell	$P_L + P_B$	50.2	57.3
Bottom Cover Plate	$P_L + P_B$	42.2	62.7
Lead Plug Top Casing Plate	$P_L + P_B$	49.7	58.0
Top Outer Cover Plate	$P_L + P_B$	55.4	58.0

* P_L = Local Member Stress; P_B = Bending Stress

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Table 6 (continued)
Structural Analysis – Maximum Calculated Stresses for all other components

ISFSI Component	Stress Type	Calculated Maximum Stress, ksi	Level D Allowable Stress, ksi
DSC Basket Assembly			
Basket	$P_L + P_B$	38.1	57.0
Rails	$P_L + P_B$	31.7	57.1
Rail Stud	Shear	17.6	19.7
Transfer Cask			
Shell	$P_L + P_B$	42.0	70.0
Top Cover Plate	$P_L + P_B$	38.2	64.4
Bottom Cover Plate	$P_L + P_B$	47.8	64.4
Hydraulic Ram Load Normal – 23.7K	$P_L + P_B$	5.8	28.4 (Level A)
Hydraulic Ram Load DSC Jamming - 95 K	$P_L + P_B$	23.2	28.4 (Level B)
Lifting Mechanisms			
	Stress Type	Maximum Calculated Stress, ksi	Code Allowable, Stress, ksi
Upper Trunnion (ANSI N14.6 Code)	Stress Intensity	7.0	13.1
Lower Trunnion (ANSI N14.6 Code)	Stress Intensity	2.51	30.5
Lifting Yoke Assembly: (ASME Code NF)	Load	Maximum Stress, ksi	Allowable Stress, ksi
Lift Beam	Ultimate Load	111.4	116.0
	Yield Load	66.9	104.0
Pin	Ultimate Load	132.9	135.0
	Yield Load	79.8	105.0
Lifting Hook	Ultimate Load	114.4	116.0
	Yield Load	103.9	104.0
	Test Load	66.8	104.0

The access road was previously evaluated for a 140-ton load. The transfer trailer load associated with moving a fuel-loaded NUHOMS-32P DSC is 131.3 ton. The calculation is based on 132.25 ton.

USAR Sections Reviewed:

All chapters of the CCNPP ISFSI USAR were reviewed.

Does the *proposed activity:*
(1.) Result in more than a minimal increase in frequency of occurrence of an accident previously evaluated in the USAR?
NO

Justification:

Accidents analyzed for the Calvert Cliffs ISFSI, for the use of NUHOMS-24P DSCs, are discussed in Section 8.2 of the USAR. Analyses of the same accidents with the use of NUHOM-32P DSCs are discussed in the new USAR Chapter 12 as follows:

12.8.4.1 Loss of Air Outlet Shielding
12.8.4.2 Tornado Winds/Tornado Missile
12.8.4.3 Earthquake
12.8.4.4 Flood
12.8.4.5 Transfer Cask Drop
12.8.4.6 Lightning
12.8.4.7 Blockage of Air Inlets and Outlets
12.8.4.8 Dry Shielded Canister Leakage
12.8.4.9 Accidental Pressurization of Dry Shielded Canister
12.8.4.10 Forest Fire
12.8.4.11 Liquefied Natural Gas Plant or Pipeline Spill or Explosion
12.8.4.12 Load Combinations
12.8.4.13 Other Event Considerations

The use of NUHOMS-32P DSC does not modify the external configuration of the DSC envelope. The interface between the DSC and the HSM during ISFSI operations and interim storage of the DSC remains unaffected except for the increased weight of the DSC, higher radiation sources and the higher heat load in the DSC.

Of the above listed events, those potentially impacted by the use of NUHOMS-32P DSC are the transfer cask drop, DSC leakage, accidental pressurization of the DSC, and load combinations. The frequency of occurrence of the other events are dependent upon natural phenomena and are independent of the type of DSC used.

<u>TRANSFER CASK DROP</u>

The cask drop accident is postulated to occur during the DSC's transit to and from the plant and the HSM. Outside dimensions of the DSC are not being altered and aspects of the postulated event, such as the height of the drop, angle of drop, surface of drop as a result are not impacted. Therefore there are no aspects of the proposed change that could potentially impact the frequency of the drop accident. There are other aspects of the proposed change such as the increases in DSC weight, source term and surface temperature which could potentially affect the consequences of the cask drop accident. These aspects of the proposed change are discussed under Question No. 3. Therefore, it is concluded that the activity does not result in more than a minimal increase in the frequency of occurrence of the cask drop accident.

DSC LEAKAGE

The NUHOMS-32P DSC shell and cover designs have been modified by moving the vent and siphon ports from shell wall to the shield plug to improve the welding operation of the shield plug to the DSC shell as compared to the NUHOMS -24P DSC shell and cover designs. This proposed change, however, has no impact on the frequency of occurrence of DSC leakage since the welding operation remains essentially the same and testing to verify leak-tightness is performed using similar techniques and equipment as used on the 24P DSC.

ACCIDENTAL PRESSURIZATION OF DSC

Accidental pressurization of DSC has been analyzed for the following two scenarios:

1. DSC in storage in the HSM with ambient temperature of 103°F, the air outlet/inlet vents blocked for up to 36 hours, and fission gases leaked out of the cladding.
2. DSC in transit at an ambient temperature of 103°F, and DSC accidentally dropped causing the fission gases to be leaked out of the cladding.

The frequency of scenario 1 does not change with the 32P DSC loaded in the HSM. Although the higher heat load of the 32P DSC design results in increased shell temperatures earlier than those experienced by the 24P design, fuel clad temperature limits are not exceed up to the 36 hour duration analyzed for this accident. Technical Specification surveillance requirements ensure that within 24 hours, that a blocked vent can be detected and actions can be taken to restore cooling.

Similarly, the frequency of scenario 2 does not change with the 32P in transit to the HSM with ambient temperature of 103°F. As discussed previously, outside dimensions of the DSC are not being altered and aspects of the transfer cask drop event, such as the height of the drop, angle of drop, surface of the drop are not impacted. As a result, there are no aspects of the proposed change that could potentially impact the frequency of the drop accident. There are other aspects of the proposed change such as the increases in DSC weight, source term and surface temperature which could affect the consequences of the cask drop accident. These aspects of the proposed change are discussed under Question No. 3. Therefore, it is concluded that the activity does not result in more than a minimal increase in the frequency of occurrence of the accidental pressurization of the DSC accident.

LOAD COMBINATIONS

With exceptions noted under Question 7 of this evaluation, the load combinations are otherwise the same for the NUHOMS-32P and 24P DSC systems for all specified normal, off-normal, and postulated accident conditions. For all load combinations, the design acceptance criteria for the NUHOMS-32P and 24P DSC shells, transfer cask, lifting trunnions, HSM, and other miscellaneous components are the same. The NUHOMS-32P DSC internal basket assembly is designed to meet the ASME code requirements applicable for core support structures, similar to the NUHOMS-24P DSC basket assembly. The same transfer cask and equipment are used to transport the NUHOMS-32P and NUHOMS-24P DSC from the spent fuel storage pool to the ISFSI, and the cask travel path and the bounding ambient temperature range (-3°F and 103°F) are the same while transporting either fuel-loaded DSC. Load combinations structural reanalysis for the NUHOMS-32P DSC, transfer cask, and HSM which account for the effects of the additional 8 fuel assemblies, increased fuel assembly weight, and basket design, confirm that the stresses imposed by the NUHOMS-32P DSC on all important-to-safety components are maintained within the required structural limits for the specified normal, off-normal, and postulated accident conditions (Reference Series 2, and 7). Therefore, the frequency of occurrence of an accident will not be increased by the NUHOMS-32P DSC load combinations.

HSM EVALUATION

Structural reanalysis of the HSM concrete structure, DSC supports, and miscellaneous structural steel components of the HSM storing the NUHOMS-32P DSC confirm the structural integrity of the HSM under all normal, off-normal, and accident conditions (Reference Series 7). To assure that the HSM concrete design temperature is not exceeded, the HSM vents are required to be inspected at 24 hour intervals (see Technical Specifications Surveillance Requirement 4.4.1.2) to verify that they are open, and to have them unblocked within 12 hours after a blockage is discovered.

- It is noted here that the time allowed to unblock the vents is being decreased from 24 hours to 12 hours. This change is deemed acceptable, because the vents are inspected every 24 hours. If a vent is found blocked cooling can be restored within 12 hours.

Therefore the frequency of occurrence of an accident will not be increased by the decrease in time allowed to unblock vents in the HSM.

Conclusion:

In conclusion, the affects of the proposed changes do not increase the frequency of any of identified accidents discussed above and previously evaluated in the USAR. In general, incorporating the use of the 32P canister (which will hold 8 additional spent fuel assemblies) will reduce the annual number of DSC loading, transport and storage evolutions, however over the life of the ISFSI the number of loading and transfer evolutions will remain the same. In effect, this will have a null impact on the frequency of occurrence of the accidents discussed above.

Does the *proposed activity:*
(2.) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component (SSC) important to safety previously evaluated in the USAR?

NO

Justification:

The structure, system or component (SSC) associated with the ISFSI comprise the DSC and its internal basket, the TC and HSM, as described in the USAR Section 3.2. These SSCs are impacted by the proposed activity because of the activity's effect on their design parameters. The use of the NUHOMS-32P DSC, which is the primary focus of the proposed activity, requires changes to the following design parameters:

1. Boron concentration in the spent fuel pool
2. Weight of the DSC
3. Heat source in the DSC
4. Radiation source in the DSC

1. Boron Concentration

The proposed activity requires the boron concentration in the spent fuel pool to be increased from 1,800 ppm to 2,450 ppm, to control the criticality during loading/unloading and drying operations. The DSC and its internals are exposed to the increased soluble boron concentration only during cask loading (and unloading) which is expected to take less than 24 hours. There is a substantial body of industry experience (including the use of the NUHOMS-24P DSC) with exposure of aluminum and stainless steel to borated and unborated water, which confirms that stainless steel and aluminum are suitable for these conditions. Therefore, malfunction of SSCs important to safety due to increased soluble boron is not increased.

One side effect of boron in the spent fuel pool is the generation of hydrogen. Generation of hydrogen is primarily due to corrosion chemical reactions with aluminum and galvanic corrosion in the presence of spent fuel pool water. The potential for hydrogen generation in a NUHOMS-32P DSC will be different from that in a NUHOMS-24P DSC, because of the differences in aluminum surface area and the different alloys of aluminum employed in the two designs. However, the amount of hydrogen generated is not relevant because the hydrogen concentration in the DSC is sampled before there is any welding or cutting of the top shield plug. The USAR Section 5.1.1.2 states that before welding or cutting of the top shield plug begins, a small tube is inserted into the DSC vent port. A hydrogen monitor is connected to the tube to continuously sample for hydrogen gas during the top shield plug welding/cutting process. If the hydrogen concentration reaches 60% of the lower flammability limit, welding/cutting activities are stopped. The DSC air space is then purged with filtered plant air. Thus, the potential for hydrogen explosion is eliminated.

2. Weight of the DSC

The weight of a fully fuel-loaded DSC is being increased from 65.0 kips to 91.0 kips due to the additional eight (8) fuel assemblies, increased fuel assembly weight, and basket design. As discussed in Attachments 1 and 5, all SSCs have been analyzed for normal, off-normal and accident conditions using the increased weight of the NUHOMS-32P DSC. The maximum resulting stresses have been shown to be smaller than the stress allowables. Therefore, the increased weight of the DSC will not result in more than a minimal increase the likelihood of occurrence of a malfunction of the SSCs.

Structural analyses for the NUHOMS-32P DSC, transfer cask, and HSM, which account for the effects of increased DSC weight, confirm that the stresses in these components remain within the allowables.

Other major equipment affected by the weight increase is the DSC transport equipment and hydraulic ram system. The transport equipment have been analyzed and found to be adequate to handle the extra load.

The Hydraulic Ram System (HRS) capacity is 80 kips. The normal force required to insert or extract a NUHOMS-32P DSC is nominally calculated to be 4.5 kips, which is well within the HRS capacity.

However, for NUHOMS-24P DSC the HRS design basis was to have a capacity equal to the DSC weight, to cope with a jammed DSC. For NUHOMS-32P this design basis will not be met. However, the existing HRS capacity of 80 kips is considered sufficient to cope with a jammed NUHOMS-32P DSC due to the following:

- ISFSI operating procedure limits the HRS force to be applied for DSC insertion to approximately 20 kips. Administrative controls are imposed when the force required exceeds the above value. The administrative controls will ensure that the force applied for insertion will not be large enough that the force required to extract a jammed DSC will be within the HRS capacity.
- NRC has accepted a HRS capacity of 80 kips for the General License NUHOMS-32PT series DSCs, whose dry weights range from 88 kips to 101.42 kips. NUHOMS-32P DSC has a dry weight of 91 kips, which is within the range of 32PT series weights.

Absent any regulatory requirements that define a minimum static coefficient of friction, while meeting all Code requirements, the NEI Guidance is met by the proposed activity and a finding of "no more than minimal increase" is appropriate for the 32P DSC design.

3. Heat Source in the DSC

The proposed activity increases the heat source in the DSC from 15.84 kW to 21.12 kW (0.66 kW/Assembly). The DSC design has been modified to provide more heat transfer surfaces to compensate for the higher heat sources. Thermal analyses for the NUHOMS-32P DSC show that the increased heat source does not increase component temperatures in the HSM, DSC, or TC beyond their design limits (Reference Series 3, 5, and 6) for normal, off-normal, or accident conditions. Therefore, the proposed activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component (SSC) important to safety previously evaluated in the USAR. Another effect of higher heat source in the DSC is a higher internal gas pressure. The DSC accident pressurization analysis for the NUHOMS-32P DSC shows that the increased number of fuel rods, higher heat source, and the decreased DSC internal volume does not increase the internal DSC pressure beyond its 100 psig design limit (Reference Series 2 and 7) for normal, off-normal, or accident conditions. Therefore, the likelihood of occurrence of a malfunction of an SSC important to safety due to increased heat source in a NUHOMES-32P DSC is not more than minimally increased.

To assure that the HSM concrete design temperature is not exceeded, the HSM vents are required to be inspected at 24 hour intervals, and if the vents are blocked then 12 hours are allowed to restore the normal air flow. Chapter 8.2.7.3 of the NUHOMS-24P USAR discusses the blocked vent accident dose consequences and provides the only USAR reference to an assumed time for a blocked vent recovery operation. The assumed time to unblock the vents is 8 hours. Since the proposed change does not affect the physical characteristics of the HSM, nor is the frequency of the accident changed as a result of the NUHOMES-32P design, the only affect of the proposed activity is an increase in the heat-up rate of the HSM concrete due to the greater heat load of the NUHOMS-32P design. The analyses discussed in Attachment 3 demonstrate that concrete temperatures remain below allowable thermal limits for up to 36 hours following a blocked vent accident. The maximum time that a blocked vent accident could go undetected is 24 hours based on Technical Specification surveillance requirements. Accordingly, the 12 hours allowed for unblocking HSM vents where the HSM contains a NUHOMS-32P DSC is within previously assumed response times for the NUHOMS-24P DSC design described in the USAR for a blocked vent accident. Dose consequences for this accident are discussed under Question 3 of this evaluation.

Accordingly, the likelihood of occurrence of a malfunction of an SSC important to safety due to increased heat load in a NUHOMS-32P DSC is not more than minimally increased.

4. Radiation Source in the DSC

The proposed activity increases the radiation (neutron and gamma) source in the DSC. The fuel assembly neutron source design basis was increased for calculating the radiation dose, in order to account for higher burnups in lower enriched assemblies. This impacts the radiation dose to the DSC components and components in the DSC vicinity. Radiation dose analyses have been performed for the DSC components. The accumulated dose is not large enough to impact any of the components. Also the confinement of a higher source of radioactive gases is not a problem, because the DSC confinement is maintained.

Shielding is provided by the DSC, transfer cask, and horizontal storage module (HSM). These SSCs have been evaluated for the NUHOMS-24P DSC in the USAR. These components are passive, and the safety requirement with regard to shielding is that they retain their integrity under the Hypothetical Accident Conditions (HAC) and normal conditions.

Shielding analyses for the NUHOMS-32P DSC, which account for the additional eight (8) fuel assemblies and basket redesign, show that the dose rates are higher on the outside of the DSC and transfer cask; but, as discussed above, the total accumulated dose values are well within the capabilities of the materials (stainless steel, aluminum, and concrete). Therefore, the likelihood of malfunction of shielding SSC important to safety due to radiation is not more than minimally increased.

The top shield plug (TSP) is redesigned in anticipation of future licensing of the NUHOMS-32P DSC for transportation in a 10 CFR 71 approved overpack. The design change involves a reduction in the lead shield thickness (decrease from 4.375 inches to 4.00 inches, minimum) and an increase in the steel plate thickness of the top shield plug top casing plate from 0.375 inches to 0.75 inches. The effect of this change is to slightly increase dose rates near the TSP (~6% higher than if the old TSP design had been used for the 32P DSC), but the effects on dose rates outside the NUHOMS HSM are insignificant (Reference Series 4). The specific dose rates are provided in Table 3. Thus the functionality of the top shield plug is essentially unchanged by this modification. Therefore, the likelihood of occurrence of a malfunction of shielding on an SSC important to safety is not more than minimally increased due to reduced lead and increased steel thicknesses in the TSP plates.

The change involves substitution of one component for another (a NUHOMS-32P DSC for a NUHOMS-24P DSC). Both of these components (the DSCs) have the same functions and analyses have shown that the functional requirements of the DSC continue to be met.

HSM EVALUATION

Structural reanalysis of the HSM concrete structure, DSC supports, and miscellaneous structural steel components of the HSM storing the NUHOMS-32P DSC confirm the structural integrity of the HSM under all normal, off-normal, and accident conditions (Reference Series 7). To assure that the HSM concrete design temperature is not exceeded, the HSM vents are required to be inspected at 24 hour intervals (see Technical Specifications Surveillance Requirement 4.4.1.2) to verify that they are open, and to have them unblocked within 12 hours after a blockage is discovered.

Therefore, likelihood of occurrence of a malfunction of a structure, system or component (SSC) important to safety previously evaluated in the USAR is not more than minimally increased.

Based on the above discussions, the proposed activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an ITS SSC previously evaluated in the USAR.

Does the *proposed activity:*
(3.) Result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR?
NO

Justification:

Accidents analyzed for the Calvert Cliffs ISFSI, for the use of NUHOMS-24P DSCs, are discussed in Section 8.2 of the USAR. Analyses of the same accidents with the use of NUHOM-32P DSCs are discussed in the new USAR Chapter 12 as follows:

12.8.4.1 Loss of Air Outlet Shielding
12.8.4.2 Tornado Winds/Tornado Missile
12.8.4.3 Earthquake
12.8.4.4 Flood
12.8.4.5 Transfer Cask Drop
12.8.4.6 Lightning
12.8.4.7 Blockage of Air Inlets and Outlets
12.8.4.8 Dry Shielded Canister Leakage
12.8.4.9 Accidental Pressurization of Dry Shielded Canister
12.8.4.10 Forest Fire
12.8.4.11 Liquefied Natural Gas Plant or Pipeline Spill or Explosion
12.8.4.12 Load Combinations
12.8.4.13 Other Event Considerations

The use of NUHOMS-32P DSCs does not modify the external configuration of the DSC envelope. The interface between the DSC and the HSM during ISFSI operations and interim storage of the DSC remains unaffected except for the increased weight of the DSC, higher radiation sources and the higher heat load in the DSC.

Of the above listed events, those potentially impacted by the use of NUHOMS-32P DSC are the transfer cask drop, DSC leakage, and Forest Fire. USAR Sections 8.1.2.1 and 8.1.2.2 discuss two off-normal events involving a jammed DSC and thermal loads. These events, however, do not have radiological consequences and as a result are not discussed under this evaluation.

Consequences of an accident refer to the accident radiological dose consequences. SSCs and their design functions related to radiological dose consequences of an accident are discussed below.

TRANSFER CASK EVALUATION

The transfer cask drop accident bounds evaluated accidents with regard to the transfer cask. Structural reanalysis of the transfer cask with a fuel-loaded NUHOMS-32P DSC confirm the structural integrity of the transfer cask under various cask drop accidents (Reference Series 7). Dose calculations for the NUHOMS-32P DSC transfer cask drop accident assume that the neutron shielding is lost. As with the NUHOMS-24P DSC, the possibility of radiation source redistribution within the DSC during the accident is conservatively accounted for by the assumption of homogenized source within the DSC. The analyses indicate that the doses are 1518 mrem/hr at 1" from the side of the Transfer Cask, and 127.6 mrem/hr at 15 feet (Reference Series 4). The latter yields an eight-hour recovery dose of 1,021 mrem at 15 feet. Section 4.7.3.3 of the USAR indicates a contact dose rate limit of 5,000 mrem/hr for this accident, and 10 CFR 72.106 establishes an accident dose limit of 5 rem at the site boundary. Section 8.2.5.3 of the USAR indicates that the NUHOMS-24P DSC contact dose rate is 1,126 mrem/hr, and the eight-hour recovery dose to an on-site worker at 15 feet will be 776 mrem. Since total dose at 15 feet is less than 5 rem, the 10CFR 72.106 (site boundary) limit is also met. Both the dose rates, and the accident dose for the NUHOMS-32P DSC with the increased neutron source term meet the limits.

Guidance contained in NEI 96-07, Appendix B, provides that, "An increase in accident consequences from a proposed activity is defined to be no more than minimal if the increase is less than or equal to 10 percent of the difference between the current bounding calculated dose value and the regulatory limit (10 CFR 72.106, as applicable)." The difference in the 32P - 15 foot worker/site-boundary dose (i.e., 1021 mrem) and the 24P - 15 foot worker/site-boundary dose (i.e., 776 mrem) is 245 mrem. Ten percent of the difference between the regulatory limit and the current bounding calculated dose value (i.e., 10% x 5,000-776) is 422 mrem. Since 245 mrem is less than the current bounding calculated dose value of 422 mrem the results meet the standard of NEI 96-07, Appendix B for being considered not more than a minimal increase.

The cask drop accident bounds all other scenarios with regard to creating conditions under which nuclear criticality is possible. Criticality is prevented (therefore the dose is limited) during dry conditions by excluding the possibility of introducing moderator (e.g., water) into the DSC cavity during the dry operations of transfer and storage. Therefore, the criticality control function of the DSC is to preclude the introduction of moderator, (e.g., water) into the interior of the DSC after it is removed from the spent fuel pool. Analyses have been performed which confirm that the NUHOMS-32P DSC will maintain its integrity under the same normal and accident conditions as the NUHOMS-24P DSC (Reference Series 1).

There are no design basis accidents involving a wet canister; however, the criticality analyses for the cask drop accident assume that the consequences of the drop result in the fuel pins being rearranged into the most reactive geometry for both wet and dry conditions. These analyses confirm that the reactivity remains below the upper subcritical limit (USL) for both wet and dry conditions (Reference Series 1).

Additionally, consistent with 10 CFR 72.124, "Criteria For Nuclear Criticality Safety," both the NUHOMS-32P DSC and the NUHOMS-24P DSC are designed such that at least two unlikely, independent, and concurrent or sequential changes are required to create conditions under which nuclear criticality is possible. Both designs require the malfunction of at least two items of equipment important to safety for criticality to be possible.

Based on the above discussion, the consequences of accidents previously evaluated in the USAR do not increase with respect to nuclear criticality control.

DSC LEAKAGE

The DSC is designed to ensure no leakage, and analyses for the NUHOMS-32P DSC confirm that the results of applicable structural analyses are unchanged for normal, off-normal and accident conditions, so a breach of the confinement barriers is not credible (Reference Series 2). Nevertheless, to demonstrate the safety of the NUHOMS-24P DSC, a non-mechanistic leak of a single DSC is analyzed in Section 8.2.8 of the USAR. The postulated accident is the instantaneous release directly to the environment of the gap inventory of Kr-85 from all the fuel rods in all 24 assemblies. The original calculated skin and total body doses for the NUHOMS-24P DSC were 17.8 and 0.11 mrem, respectively. This calculation utilized 18-month operating cycle inputs for calculating the Kr-85 gap fraction using the ANSI/ANS 5.4 method.

The NUHOMS-32P DSC doses are evaluated using the original methodology but for a 24-month operating cycle. Calculation CA03902 (Reference Series 4) determines the skin and total body dose values of 60.1 and 0.36 mrem, respectively. These dose values remain well within the regulatory limits, as specified in 10 CFR 72.106, of 50 rem to the skin and 5 rem to the total body. The increase in dose values over those reported in the USAR for the 24P are 60.1 – 17.8 or 42.3 mrem to the skin and 0.36 – 0.11 or 0.25 mrem to the total body. Ten percent of the difference between the regulatory limit and the current bounding calculated dose values for skin dose (i.e., 10% x 50,000-17.8) is 4,998 mrem and total body dose (i.e., 10% x 5000 – 0.11) is 499.9. Since 60.1 is significantly less than 4998 mrem and 0.36 mrem is significantly less than 499.9 mrem it is concluded the current bounding calculated dose values for skin and total body exposure meet the standard of NEI 96-07, Appendix B for being considered not more than a minimal increase.

HSM EVALUATION

Structural reanalysis of the HSM concrete structure, DSC supports, and miscellaneous structural steel components of the HSM storing the NUHOMS-32P DSC confirm the structural integrity of the HSM under all normal, off-normal, and accident conditions (Reference Series 7). To assure that the HSM concrete design temperature is not exceeded, the HSM vents are required to be inspected at 24 hour intervals (see Technical Specifications Surveillance Requirement 4.4.1.2) to verify that they are open, and to have them unblocked within 12 hours after a blockage is discovered. Two scenarios involving the HSM that are analyzed in the USAR have potential radiological consequences; (a) blockage of the air inlets and outlets, and (b) concrete cracking and spallation due to fire.

Direct doses from the first analyzed scenario, blockage of air inlets and outlets with a NUHOMS-32P DSC in the HSM, are bounded by the doses with a NUHOMS-24P DSC installed. The dose rate at the air inlet with a NUHOMS-32P DSC is 68 mrem/hr (Table 3), which is less than the 73 mrem/hr with a NUHOMS-24P DSC installed.

For the second analyzed scenario, the spalling of concrete from the HSM in a forest fire increases from 4.5" for the 24P to 6" for the 32P (CA06629; reference series 6). CA03945 (reference series 6) indicates that a 12" reduction in concrete thickness would produce a factor of 20 increase in dose rate at the HSM surface. This translates to an increase by a factor of 3 for a 4.5" reduction in concrete thickness, and a factor of 4.5 for a 6" reduction in concrete thickness. The original design goal was that the spalling not increase dose rates to a level beyond which repair actions could be performed by conventional methods (1 rem/hr). Forest fire spalling would result in a dose rate of 45 mrem/hr at the 32P HSM surface (4.5 x 9.9 mrem/hr from Table 3) which is still significantly below the 1 rem/hr design goal for spalling repair. Thus, it can be concluded that the 32P DSC would also not adversely impact the ability to repair spalled concrete following a forest fire. In addition, the impact of forest fire induced spalling must be evaluated against the requirements of 10CFR72.106, which establishes an accident dose limit of 5 rem at the site boundary. ISFSI USAR Section 2.1.2.2 indicates that the minimum distances from the ISFSI to the site boundary is 3900 feet. From the results of CA06293 and CA06058 the maximum dose rate at 3000 feet from an ISFSI site consisting of 120 loaded HSMs of 32Ps or 24Ps is less than 1E-5 mrem/hr in both the N/S and E/W directions. A conservative normal annual site boundary dose would then be 0.09 mrem (1E-5 mrem/hr x 8760 hrs) for either a fully loaded 24P or 32P ISFSI site. While concrete spalling caused by a forest fire would not be allowed to persist for a year without repair, such spalling would conservatively increase the annual site boundary dose to 0.27 mrem for a full 24P ISFSI site (3 x 0.09 mrem), and 0.41 mrem for a full 32P ISFSI site (4.5 x 0.09 mrem). Thus, the change to a 32P conservatively increases the annual site boundary dose following a forest fire by 0.14 mrem (0.41 - 0.27). Ten percent of the difference between the regulatory limit and the current bounding calculated dose value (i.e., 10% x 5,000-0.27) is 499 mrem. Therefore, these results meet the standard of NEI 96-07, Appendix B for being considered a minimal increase.

A third accident scenario for the HSM presented in the NUHOMS Topical Report with radiological consequences, "Loss of Air Outlet Shielding," is evaluated for the CCNPP NUHOMS-24P in Section 8.2.1 of the ISFSI USAR. This accident was considered not credible for CCNPP HSM because the air outlet shielding is designed to remain in place and withstand all design events including the effects of tornado missiles. The HSM design is not being altered for use with the 32P, and therefore, this conclusion still applies.

Based on the above discussion, the proposed activity does not exceed the accident dose limits established by 10 CFR 72.106 and does not produce more than a minimal increase in the consequences of an accident previously evaluated in the ISFSI USAR

Does the *proposed activity:*
(4.) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the USAR?
NO

Justification:

The structure, system or component (SSC) associated with the ISFSI comprise the DSC and its internal basket, the TC and HSM, as described in the USAR Section 3.2. Of these SSCs, only the DSC and its internal basket assembly have been changed. Structural evaluations of the DSC and its internal basket assembly confirm their structural integrity under all specified normal, off-normal, and accident conditions (Reference Series 2). Structural reevaluation of the transfer cask, the reinforced concrete HSM, and the DSC support structure for the increased weight and thermal load of the NUHOMS-32P DSC confirm the structural integrity of these components under all specified normal, off-normal, and accident conditions (Reference Series 2). No changes to the reinforced concrete HSM and its DSC support structure, and the transfer cask have been made.

CRITICALITY CONTROL

Consistent with 10 CFR 72.124, "Criteria For Nuclear Criticality Safety," both the NUHOMS-32P DSC and the NUHOMS-24P DSC are designed such that at least two unlikely, independent, and concurrent or sequential changes are required to create conditions under which nuclear criticality is possible. Therefore, the consequences of a malfunction of any single piece of equipment important to criticality safety are unchanged.

DSC LEAKAGE DOSE

The SSCs important to safety with regard to radiation safety (doses to the operators and doses to the public) consists of the DSC, transfer cask, and horizontal storage module. The DSC is designed to ensure no leakage, and the analysis for the NUHOMS-32P DSC confirm that a breach of the confinement barriers is not credible. A non-mechanistic leakage of the DSC is discussed in response to Question 3 (above) regarding the consequences of an accident.

Thus, the consequences of a malfunction of an SSC important to safety previously evaluated in the USAR remains unchanged.

Does the *proposed activity:*
(5.) Create a possibility for an accident of a different type than any previously evaluated in the USAR?
NO

Justification:

Accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the USAR. They consist of loss of shielding, external missiles, earthquake, flood, cask drop, lightning, blockage of air inlets and outlets, DSC leakage, DSC overpressurization, forest fire, liquefied natural gas plant or pipeline spill or explosion, load combinations, and the onsite storage of flammable liquid fuel.

There is no change to the design of the NUHOMS system caused by this activity other than the use of the NUHOMS-32P DSCs. There is, however, a change in operation in that soluble boron is credited for criticality control. This activity does not modify the external configuration of the DSC envelope. The interface between the DSC and the HSM during ISFSI operations and interim storage of the DSC remains unaffected except for the increased weight of the DSC and the additional heat load in the DSC. The HSM vents are required to be inspected at 24 hour intervals to verify that they are unblocked, and to assure the HSM concrete short term local design temperature limit of 395°F is not exceeded.

The changes that support the upgrade from the NUHOMS-24P DSC to the NUHOMS-32P DSC are:

- Redesign of the DSC basket to hold eight (8) more fuel assemblies
- Use of fixed neutron absorber plates in the DSC to control criticality
- Increased fuel pool soluble boron concentration from 1,800 ppm to 2,450 ppm
- Removing the requirement for limiting initial enrichment based on burnup
- Increased fuel assembly neutron source term
- Modified top shield plug
- Increased fuel assembly weight

The components of the NUHOMS-32P DSC and NUHOMS-24P DSC internals perform passive safety functions, which they fulfill by retaining their structural integrity and remaining in place. Calculations for the NUHOMS-32P DSC basket confirm that the results of applicable structural analyses are unchanged for Hypothetical Accident Conditions and normal operation, so the guide sleeves, basket plates, and outer rails and aluminum inserts remain in place. These analyses also confirm that the DSC shell remains intact and precludes the release of radioactive material, or the ingress of water (moderator). Therefore, there is no accident of a different type than analyzed in the USAR, that is created by the redesign of the basket to hold eight (8) more assemblies (Reference Series 2 and 7).

The fixed neutron absorbers are borated aluminum plates between two unborated aluminum plates. The effectiveness of borated aluminum plates as neutron shielding is established through testing (Reference 14). In addition, the aluminum and stainless steel plates are not adversely affected by exposure to borated and unborated water as confirmed by a substantial body of industry experience. Therefore, the possibility for an accident of a different type than any previously evaluated in the USAR involving the use of fixed neutron absorbers is not credible.

As discussed in Attachment 6, one side effect of boron in the spent fuel pool is the generation of hydrogen. Generation of hydrogen is primarily due to corrosion chemical reactions with aluminum and galvanic corrosion in the presence of spent fuel pool water. The potential for hydrogen generation in a NUHOMS-32P DSC will be different from that in a NUHOMS-24P DSC, because of the differences in aluminum surface area and the different alloys of aluminum employed in the two designs. However, the amount of hydrogen generated is not relevant because the hydrogen concentration in the DSC is sampled before there is any welding or cutting of the top shield plug. The USAR Section 5.1.1.2 states that before welding or cutting of the top shield plug begins, a small tube is inserted into the DSC vent port. A hydrogen monitor is connected to the tube to continuously sample for hydrogen gas during the top shield plug welding/cutting process. If the hydrogen concentration reaches 60% of the lower flammability limit, welding/cutting activities is stopped. The DSC air space is then purged with filtered plant air. Thus, the potential for hydrogen explosion is eliminated.

The means for ensuring subcriticality are different for the NUHOMS-24P DSC and NUHOMS-32P DSC. The current NUHOMS-24P DSC ISFSI criticality analysis of record credits no soluble boron but does assume a fuel burnup credit, relying on a fresh fuel enrichment limit of 1.8 w/o U-235 and an equivalency calculation for higher enrichments. The presence of fixed neutron absorbers in the NUHOMS-32P DSC basket assembly along with credit for soluble boron in the moderator eliminates the need for burnup credit to meet criticality acceptance criteria.

A dilution event is considered not credible, because the spent fuel pool water is normally maintained at 2,450 or greater ppm boron concentration and because multiple alarms and operator actions exist to prevent this scenario. (An exemption request with the provisions of 10CFR50.68 was approved by the NRC to allow the use of boron credit in lieu of fuel burnup credit.) After the DSC is dried and sealed, there is no credible way for water to enter the DSC. Soluble boron is only credited in accident scenarios for the NUHOMS-24P DSC.

Increasing the neutron source term (from 2.23E+08 n/s/assy to 3.3E+08 n/s/assy) does not impact equipment important to safety. As discussed below, the increased neutron fluence remains well within the capabilities of the materials of construction, concrete, stainless steel and aluminum.

The effects of radiation on concrete from a filled DSC are determined to be negligible for the NUHOMS-24P DSC in Section 8.1.1.5 of the USAR. This is based on the evaluation performed in Section 8.1.1.5 of the NUHOMS-24P Topical Report (Reference 12). In the case of neutron irradiation, the NUHOMS 24P Topical Report determines that an estimated 50-year fluence of 1.2E+14 n/cm^2 for a source of 3.715E+09 n/sec-DSC (24 x 1.548E+08 n/sec-assembly) will not impact the structural integrity of the concrete. This conclusion remains valid for the CCNPP NUHOMS-24P DSC because the neutron source term is not significantly higher than that evaluated in the Topical Report (2.23E+08 n/sec-assembly /1.548E+08 n/sec-assembly = 1.44). In the case of the NUHOMS-32P the DSC neutron source term will be 1.056E+10 n/sec-DSC (32 x 3.3E+08 n/sec-assembly) which is a factor of approximately 2.8 higher than that used in the Topical Report. This yields an estimated 50-year fluence of 3.4E+14 n/cm^2 for the concrete, which is conservative because it does not account for the additional shielding present in the NUHOMS-32P DSC basket design. This value is still well below the critical value for neutron induced degradation of concrete, which is approximately 1.0E+19 n/cm^2 (Reference 11).

In the case of gamma irradiation, the NUHOMS-24P Topical Report (Reference 12) determines that a gamma source of 3.85E+16 MeV/sec-DSC (24 x 1.6E+15 MeV/sec-assembly) results in a gamma flux of 6.8E+10 MeV/cm^2-sec in the HSM concrete. The temperature rise in concrete due to this flux is considered negligible per ANS/ANSI 6.4-1977. In the case of the NUHOMS-32P DSC, the gamma source is higher by a factor of 1.27 (32 x 1.53E+15 MeV/sec-assembly =

4.9E+16 MeV/sec-DSC). However, as the data in Table 3 shows, even with the increase gamma source in the NUHOMS-32P DSC, the gamma dose rates outside the DSC are comparable or lower than the NUHOMS-24P DSC due to the increased shielding present in the NUHOMS-32P DSC basket design. The lower dose rates are due to reduced gamma flux on the outer surface of the DSC, which is the flux that impacts the HSM concrete. Therefore, since the incident gamma flux will be the same or lower than the NUHOMS-24P DSC, the NUHOMS-32P DSC gamma flux will also have a negligible impact on the concrete temperatures of the HSM. Hence, the effects of radiation on the HSM concrete are determined to be negligible for a NUHOMS-32P DSC inside the HSM.

The top shield plug (TSP) was redesigned in anticipation of future licensing of the NUHOMS-32P DSC for transportation in a 10 CFR 71 approved overpack. The design change involves a reduction in the lead shield thickness (decrease from 4.375 inches to 4.00 inches, minimum) and an increase in the steel plate thickness of the top shield plug top casing plate from 0.375 inches to 0.75 inches. The specific dose rates are provided in Table 3. The functionality of the top shield plug is essentially unchanged by this modification so it does not create the possibility of an accident of a different type than any previously evaluated in the USAR.

Thermal analyses for the NUHOMS-32P DSC which account for the additional eight fuel assemblies show that the increased decay heat load does not increase component temperatures in the HSM, DSC, or TC beyond their design limits (Reference Series 3, 5, and 6) for normal, off-normal, or accident conditions. Therefore, this modification does not create the possibility of an accident of a different type than any previously evaluated in the USAR.

The DSC accidental pressurization analysis for the NUHOMS-32P DSC which accounts for the additional eight fuel assemblies shows that the internal DSC pressure is not increased beyond its 100 psig design limit (Reference Series 2 and 7) for normal, off-normal, or accident conditions. Therefore, this modification does not create the possibility of an accident of a different type than any previously evaluated in the USAR.

The change involves the substitution of one component for another (a NUHOMS-32P DSC for a NUHOMS-24P DSC). Based on the discussion above, the proposed activity does not create a possibility for an accident of a different type than any previously evaluated in the USAR.

Does the *proposed activity:*
(6.) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR?
NO

Justification:

Section 3.2 "STRUCTURAL AND MECHANICAL SAFETY CRITERIA" of the USAR states that the Calvert Cliffs ISFSI components that are important to safety are the reinforced concrete HSM and its DSC support structure, the DSC and its internal basket assembly, and the transfer cask. The proposed activity directly impacts the DSC and its internal basket, which are redesigned to store 32 fuel assemblies as opposed to 24 fuel assemblies stored in NUHOMS-24P DSC. The increase weight and heat and radiation source of the new DSC then indirectly affects the loads and stresses on the TC and HSM.

Detailed evaluations of the SSCs have been performed for normal and off-normal operating conditions and accident conditions, as reported in Attachment 3. Results of the analyses have shown that the proposed activity will not result in any malfunction of the DSCs.

CRITICALITY CONTROL

Structural analysis of the new basket assembly has shown that the basket assembly will be able to maintain its configuration even under accident conditions. Therefore, criticality control will be maintained.

FUEL ASSEMBLY SPACING

The proposed activity requires the DSC guide sleeves to be reduced in size from 8.7" to 8.5" and the spacing between them to be reduced from 10.36" to 9.125". This change has the potential to impact criticality control within the DSC. The DSC design has been modified by adding borated aluminum plates between the fuel assemblies (between the guide sleeves) to compensate for the reduction in spacing. The effectiveness of borated aluminum plates as neutron shielding is established through testing (Reference 14).

Consistent with 10 CFR 72.124, "Criteria for Nuclear Criticality Safety," the continued efficacy of the neutron absorber material, borated aluminum, when exposed to borated water in the spent fuel pool is confirmed by industry and CCNPP experience. The duration of exposure to borated water is normally less than 24 hours. There is a substantial body of industry and CCNPP experience with exposure of aluminum to borated and unborated water, which confirms that stainless steel and aluminum are suitable for these conditions. Therefore, the malfunction of the borated aluminum neutron absorbers is not credible.

As discussed in Attachment 3, the NUHOMS-32P DSC configuration has been analyzed for criticality during normal, off-normal, and accident conditions, the analysis results show that criticality meets the applicable criteria, and hence does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR.

SHIELDING & DOSE

The results of postulated failures to the NUHOMS-32P DSC have been evaluated for the effects of the upgrade. These analyses have confirmed that upgrading to the NUHOMS-32P DSC does not cause materials of construction to be challenged, or design basis limits to be exceeded. Therefore there are no new failure modes or mechanisms that would change the results of postulated and hypothetical malfunctions previously evaluated in the USAR.

The change involves the substitution of one component for another (a NUHOMS-32P DSC for a NUHOMS-24P DSC). Based on the discussion above, the proposed activity will not create the possibility of a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR.

Does the *proposed activity:*
(7.) Result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered?
YES.

Justification:

The fission product barriers for a spent fuel storage cask system and their related parameters are tabulated below.

Barrier	Function	Design Parameter	Current Limit	Proposed Limit	Exceeded/ Altered?
Fuel Cladding	Protect Against Gross Rupture	Cladding Temp	635 °F / 1058°F	635 °F/ 1058°F	No / No
		Criticality (K_{eff})	<0.95	<0.95	No
		Decay Heat	0.66 kW/Assembly	0.66 kW/Assembly	No
Confinement Boundary (DSC Shell)	Keep Fission Products Confined	Canister Accident Design Pressure	50 psig	100 psig	Yes (Altered)
		Stress Allowable	64 ksi	57.3 ksi	Yes (Altered)
		Leak Rate (Helium)	$<10^{-4}$ atm-cc/s	$<10^{-4}$ atm-cc/s	No

Following are details of the above parameters.

CLADDING TEMPERATURE

The fuel cladding damage threshold temperature at CCNPP for normal operating conditions is 635°F (335°C) (Reference 18). For short-term off-normal, short-term accident, and fuel transfer operations (e.g. vacuum drying of the cask) the fuel cladding temperature limit is 1,058°F (570°C) per Reference 18.

The fuel cladding temperature has been evaluated for various operating conditions and ambient temperatures, and the results are tabulated in Attachment 3, Table 4. It is seen that for normal storage conditions the cladding is maintained at a temperature not exceeding 597°F (below 635°F), and during off-normal and accident conditions the cladding temperature does not exceed 838°F (below 1,058°F).

CRITICALITY

Design basis criticality analyses are performed for Combustion Engineering (CE) design 14x14 non-Value Added Pellet (VAP) fuel assemblies containing UO_2 enriched up to 4.5 wt% U-235. Loading of the fresh fuel in NUHOMS-32P was considered, same as is done for NUHOMS-24P analysis. The Upper Subcriticality Limit (USL), as defined in NUREG/CR-6361, Section 4, is calculated for the CCNPP fuel and NUHOMS-32P design through a series of calculations to be 0.9422. An effective multiplication factor (k_{eff}) that is less than USL ensures that the k_{eff} will be lower than the regulatory limit of 0.95 even when the bias and uncertainties in design parameters are taken into account.

The criticality within the NUHOMS-32P DSC is analyzed for the following conditions:

- Normal operations
  - wet loading (optimum moderation)
  - dry storage (via moderator exclusion)
- Off-normal or accident conditions
  - fuel mis-load. Fuel mis-load analyses are performed for CE design 14x14 Value Added Pellet (VAP) fuel assemblies containing UO_2 enriched up to 5.0 wt% U-235, to determine how many VAP assemblies can be loaded without exceeding the k_{eff} limit.
  - off-normal poison plate thickness and accidental cask drop.

The criticality analyses are documented in Reference Series 1. The fuel mis-load analysis demonstrates that up to two VAP assemblies can be loaded at optimum moderation without exceeding the k_{eff} limit. These analyses demonstrate that the k_{eff} in all cases is below the USL of 0.9422. The analyses confirm that the Calvert Cliffs site-specific NUHOMS-32P design satisfies the requirements of 10 CFR 72.124 for normal, off-normal, and hypothetical accident conditions.

CANISTER DESIGN PRESSURE

The pressure design limit for NUHOMS-32P is 100 psig, which is higher than the pressure design limit for NUHOMS-24P of 50 psig. NUHOMS-32P canister is analyzed for a pressure of 100 psig, and is found to be capable of withstanding it without exceeding any of the stress allowables. However, this requires a prior NRC approval, because the design basis limit for a fission product barrier is altered.

STRESS ALLOWABLE

The NUHOMS-32P DSC confinement boundary consists of the DSC shell, top shield plug top casing plate, siphon/vent block, alignment block, top shield plug lifting lug round bar, bottom cover plate, and the associated structural joint welds. The design requirements for the NUHOMS-32P and NUHOMS-24P DSC shell assembly and the associated structural welds are the same. Stress analyses of the NUHOMS-32P DSC for the effects of loads associated with dead weight, pressure, thermal, handling, seismic, and cask drop confirm that stresses in the DSC shell assembly components and the associated structural welds are within the current limits and the structural integrity of the DSC confinement boundary is maintained under all specified normal, and off-normal conditions except for the 100 psig applied to the outer pressure boundary, and the accident condition for 100 psig applied to the inner pressure boundary (reference series 2).

Confinement boundary stress allowables have been altered for one off-normal and most accident conditions as follows (reference series 2, CA06359, Rev. 5, pp34 and A7):

1. The condition of 100 psig applied to the outer pressure boundary (outer top cover plate) has been changed from an off-normal to an accident condition; correspondingly, the stress limits for this condition have been changed from ASME service level C to ASME service level D allowables.

2. For all accident conditions except 100 psig applied to the inner pressure boundary, the analysis has been changed from an elastic to an elastic/plastic analysis, with a corresponding change from ASME elastic allowables to ASME elastic/plastic allowables.

In conclusion, two of the design basis limits for a fission product barrier, namely the DSC design pressure and the confinement boundary stress allowables, are altered. Prior NRC approval is required. The change in design pressure was submitted to the NRC as part of the license amendment application, and the analysis which include the change in stress allowables were sent to the NRC with the response to RAI No. 1 for that application.

Does the *proposed activity:*
(8.) Result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses?
YES

Justification:

Analyses performed for this activity consist of thermal, criticality, radiation dose and shielding and structural analyses. As demonstrated below, except in the cases of changes to methodologies used for crediting burnup to fixed and dissolved neutron absorber credit for the criticality analysis and the analysis changes from an elastic to an elastic/plastic analysis for all accident conditions except 100 psig applied to the inner pressure boundary, as discussed under previous Question No. 7, the analytical methods used are either the same, or where new methods are used they are approved by the NRC for the intended application, that is, for a NUHOMS dry storage system. Where elements of an analytical method have changed, e.g., revisions to an analytical code, they yield results that are essentially the same or are conservative. In accordance with NEI 96-07, Rev. 1, Appendix B, section B.4.3.8, such changes are not departures from a method of evaluation described in the USAR..

In particular the computer codes, ANSYS, SCALE and MCNP are verified in accordance with Transnuclear Q/A computer qualification/verification procedure. The Transnuclear Q/A Program has been accepted and audited by the NRC and CEG. The analysts that utilize the Codes are trained both by attending training classes given by the Code developers and by experienced Transnuclear Code users. This meets the NRC Generic Letter 83-11 criteria.

THERMAL EVALUATION

The primary portion of the thermal evaluation of the NUHOMS® 32P uses a methodology that differs from the thermal analysis methodology utilized for the NUHOMS-24P as described in the CCNPP USAR. The new methodology has been compared in detail with the methodology used for the thermal analysis of the NUHOMS® CoC 1004 amendment 5 for the 32PT.

The 32P thermal methodology has three major new features shown in the following table.

Feature	CCNPP ISFSI 24P USAR	32P
Solution method	HEATING6 finite difference	ANSYS finite element
Model geometry	2D	3D
Treatment of effective transverse thermal conductivity of fuel.	Temperature at the boundary of a fuel assembly and peak temperatures are calculated using an iterative process. These temperatures are then used with the methodology in the NUHOMS®-24P Topical Report for effective fuel conductivity values. See Section 8.1.3 of the USAR.	From detailed finite element model of fuel, according to method of the TRW Spent Nuclear Fuel Effective Conductivity Report.

Use of Approved 32PT Thermal Methodology for the 32P within Appropriate Constraints and Limits in Accordance with NEI 96-07 B4.3.8.2

The new methodology has been compared in detail with the methodology used for the thermal analysis of the NUHOMS® CoC 1004 amendment 5 for the 32PT. The use of the 32PT methodology is appropriate to the CCNPP ISFSI with the 32P canister as follows:

a) The 32P maximum thermal load of 21 kW is below the 24 kW limit of the 32PT.
b) The 32P and 32PT basket designs both rely on conduction and radiation from the fuel through plates adjacent to the fuel compartments to peripheral transition rails to the canister wall, rather than tube and disc designs, in which convection is also a factor.
c) The 32PT is analyzed in the standard Model 80/102 HSM; the 32P is analyzed in the CCNPP HSM, which is similar in internal dimensions and air exit paths.
d) The 32PT is analyzed in a transfer cask with a liquid neutron shield, while the CCNPP transfer cask has a solid neutron shield. The application of the 32PT methodology to the CCNPP transfer cask is appropriate because the treatment of conduction through the solid neutron shield is much simpler than the conduction/convention heat transfer in the liquid neutron shield.

1

The following Tables 1-3 break down various parts of the thermal evaluation into more detailed constituents, compares the 32P and the 32PT approved methodology, and addresses any differences. One element of the approved 32PT methodology that differs is the 32P thermal analysis, the version of the ANSYS code used, is dealt with here separately.

ANSYS Revision Comparison

ANSYS revision 5.6 was used for the thermal analysis of the NUHOMS® system with the 32PT canister. ANSYS revisions 5.6, 5.7, and 6.0 are used to perform the 32P thermal analyses (Reference Series 3, 5, and 6).

DCALC CA06619 benchmarks revisions of ANSYS 5.6 and 6.0 to demonstrate that the results are essentially the same. Changes introduced in ANSYS 5.7 were not used in any of the 32P thermal analysis so ANSYS 5.6 and 5.7 are identical for this purpose. ANSYS 8.1 was used to evaluate temperatures with a finer homogenized fuel mesh. DCALC CA06639 benchmarks ANSYS 5.7 and 8.1 and finds no difference for this calculation.

Therefore, changes to elements of the methodology, (i.e.,) various ANSYS revisions used subsequent to ANSYS 5.6 are not departures from a methodology described in the USAR.

Table 1: Modeling of DSC Internals

Constituent	Approved 32PT	32P	comment
Basket model	Half length, 3D	Full length, 3D	Because the location of the fuel in the canister is slightly off-center axially, a full-length model represents the basket and fuel more accurately, and is therefore appropriate.
Calculation of Effective Transverse Fuel Conductivity	Transverse effective conductivity of the homogenized fuel is calculated based on 2D ANSYS model of fuel cross section. No convection is considered in the model.	No change	
Calculation of Effective Axial Fuel Conductivity	Calculated from cross-sectional area of Zircaloy cladding only	No change	
Homogenized fuel mesh density	14x14	Non-uniform mesh, courser at center of fuel, finer at perimeter	As-discussed below a bounding thermal analysis was performed for 32P using a 14x14 mesh density with acceptable results
Heat generating conditions	Heat generating conditions applied on homogenized fuel assembly regions using peaking factors.	No change	

Table 2: Modeling of DSC in Transfer Cask

Constituent	Approved 32PT	32P	comment
Transfer cask model	2D of cask cross section	Full length 3D quarter symmetry	3D required to model neutron shielding material and gussets explicitly. Because very little heat is rejected through the ends of the cask, the full length model will not significantly change results for maximum cladding and component temperatures.
Canister to cask radiation heat transfer	ANSYS /AUX 12	No change	
Canister centering	DSC shell is off set within the transfer cask. Air gaps at top and bottom are 0.702 inch and 0.108 inch respectively. DSC shell rests on the cask rails.	DSC is centered in the cask. No cask rails are considered in the model.	A study shows that considering a uniform gap is conservative regarding the maximum DSC shell temperature.
Heat input	Heat load is applied as uniform heat flux over the inner surface of the DSC. Insolence is applied as uniform heat flux over the top half of the cask outer surface using an absorptivity factor.	No change	
Cask to ambient convection	Free convection	Same, using revised convection coefficients	Revised coefficients are input only, not methodology.
Radiation from cask to ambient	Radiation to ambient is modeled using ANSYS /AUX12 methodology	Radiation is lumped into an effective convection coefficient to ambient.	Lumping convection and radiation together yields essentially the same results for maximum cladding and component temperatures as treating radiation explicitly.

Table 3: Modeling of DSC in HSM

Constituent	Approved 32PT	32P	comment
HSM model	2D of the HSM cross section at mid-length	No change	
Heat input	Heat load is applied as uniform heat flux over the inner surface of the DSC.	No change	
Convective heat removal from canister	Convection applied over DSC and HSM walls using various bulk temperatures.	No change	
Radiative heat removal from canister	ANSYS /AUX12 models radiation from canister to heat shields and heat shields to concrete	No change	
Accident analysis of blocked vents	Convection in closed cavity is considered for the accident analysis of the blocked vents.	No convection is considered for the blocked vent case analysis.	No convection yields more conservative results for the fuel cladding temperature.

Evaluation of Homogenized Fuel Mesh Size

The detailed breakdown and comparison of the 32P thermal analysis with the 32PT thermal analysis found only one area where there was a difference. The 32PT finite element model used a 14x14 uniform transverse mesh for each homogenized fuel assembly, while the 32P used a non-uniform mesh, coarser at the center of the fuel, but finer at the boundaries. In order to evaluate this difference the 32P thermal analysis for normal condition and for the blocked vent accident condition was re-run with a 14x14 transverse fuel mesh. The result was the following changes:

	Temperature increase (°F)	
	Normal Condition	Blocked Vent Accident
Fuel Cladding	+1	+1
Fuel Compartments	+9	+7
Aluminum Basket Plates	+10	+7
Stainless Steel Bars	+9	+8
Basket Rails	+4	+3

These changes resulted in the following increases in basket temperatures:

Component	Maximum Temperature for Normal Storage Conditions (°F)		
	Original Model	Temperature increase	Updated Temperatures
Fuel Compartments	571	+9	580
Aluminum Basket Plates	567	+10	577
Stainless Steel Bars	570	+9	579
Basket Rails	419	+4	423

These changes resulted in the following increases in the Maximum Temperature for Transfer Conditions at 103°F, ambient temperatures:

Component	Maximum Temperature for Transfer Conditions, 103°F Ambient (°F)		
	Original Model	Temperature increase	Updated Temperatures
Fuel Compartments	717	+9	726
Aluminum Basket Plates	717	+10	727
Stainless Steel Bars	717	+9	726
Basket Rails	573	+4	577

The basket temperature in the center region increases 10°F from 717°F to 727°F and in the periphery increases 4°F from 573°F to 577°F. The structural analysis uses a uniform temperature of 725°F for the entire basket. Since the maximum stress in the basket does not occur in the high temperature region, this increase does not affect the results of the structural analysis.

It is conservatively assumed that the temperature of the gas increases by the maximum increase for any component (10°F). This temperature increase results in an internal pressure increase of 0.9 psi. This increases the accident storage pressure from 98.5 to 99.4 psig, still within the 100 psig design limit.

Therefore, the difference in the meshing of the fuel does not have a significant effect on maximum fuel cladding temperature, structural analysis, or internal pressure.

CRITICALITY CONTROL EVALUATION

Criticality safety of the NUHOMS-32P DSC is evaluated using computer code CSAS25/KENO V.a of SCALE 4.4. This is an upgrade from the code that was used to evaluate the NUHOMS-24P DSC, namely CSAS4/KENO V.a of SCALE 3. This change will not be evaluated because the following change requires the criticality analysis to be submitted for NRC approval.

The current NUHOMS-24P DSC ISFSI criticality analysis of record credits no soluble boron but does assume a fuel burnup credit, relying on a fresh fuel enrichment limit of 1.8 w/o U-235 and an equivalency calculation for higher enrichments. The presence of fixed neutron absorbers in the NUHOMS-32P DSC basket assembly along with credit for soluble boron in the moderator eliminates the need for burnup credit to meet criticality acceptance criteria.

The ISFSI Technical Specification 3/4.2.1 (Dissolved Boron Concentration) requires 1,800 ppm of soluble boron in the spent fuel pool (SFP), which is credited by the double contingency principle for postulated accidents and abnormal conditions (e.g., optimum moderation, assembly mis-loading). The proposed NUHOMS-32P DSC ISFSI criticality analysis credits 2,450 ppm of soluble boron and no burnup credit. Soluble boron dilution is assumed not to be a credible accident. The criticality analysis for the NUHOMS-24P DSC considered an optimum moderation with pure water. That is a conservative assumption, but is considered not credible, because the spent fuel pool water is normally maintained at 2,450 or greater ppm boron concentration. After the DSC is dried and sealed, there is no credible way for water to enter the DSC.

Therefore, the use of boron credit in lieu of fuel burnup credit is considered to constitute a new method for the criticality analysis described in the ISFSI USAR. The NUHOMS-32P DSC criticality analysis was sent to the NRC in the December 12, 2003 License Amendment Request to revise the Technical Specifications to support the ISFSI NUHOMS-32P upgrade (Reference 10).

SHIELDING and DOSE EVALUATION

The methodology used for the 24P shielding evaluation in the CCNPP ISFSI USAR, sections 7.2 and 7.3, consists of calculation of source terms by the computer code ORIGEN2, and shielding analysis using the three-dimensional Monte Carlo Code MCNP4C. The MCNP4C methodology replaced the computer programs ANISN (one dimensional discrete ordinates), and SKYSHINE (empirical far-field reflection) used for the original 24P dose calculations. The use of MCNP4C has been previously accepted by the NRC and was previously evaluated for the 24P DSC in SE00160.

As documented in calculation CA06293, (Reference Series 4), the current dose rate calculations for the NUHOMS-32P DSC were prepared using MCNP4B2, with essentially the same modeling as used in the revised 24P calculations. The per-assembly source term was not recalculated for the 32P; the source term from the 24P was used, with the exception that the per-assembly neutron source strength (an input) was scaled up to 3.3E8 n/sec, as discussed in Attachment 3. RSICC Code Package CCC-700 identifies the major changes from MCNP4B to MCNP4C. These changes represent changes to inputs (e.g., ENDF/B cross-sections), features that are not used (macrobodies), or changes to elements of the method that result in essentially the same results, since the results are well within the general accuracy of shielding calculations. In addition, comparisons between 4B and 4C performed at Calvert Cliffs (included with Reference Series 4) confirm that differences in results between the same case run in 4B and 4C are statistically insignificant (differences are within the relative error of the calculation result). Comparisons between MCNP4C and an even earlier version, MCNP4A are also documented in CA05925 Appendix E and also show statistically insignificant differences in the results produced by identical cases run on both versions. Finally, MCNP 4B has also been previously accepted by the NRC (CoC No. 9293, Rev. 1 for the TN-68 Transport Package and CoC No. 9302 for the NUHOMS®-MP197 Transport Package).

Based on NRC's review and approval of the LARs using MCNP4B, and based on a review of the differences between MCNP4C and MCNP4B, the changes to elements of the methodology are not departures from a method of evaluation described in the USAR.

In addition, the calculation performed to evaluate off-site dose for a non-mechanistic DSC leakage event (USAR Section 8.2.8.3) was performed for the 32P DSC using the same methodology as used for the 24P DSC (i.e., same dose conversion factors, ANSI/ANS 5.4 calculated gap fractions, χ/Q value, etc.). Inputs were changed to reflect the increased number of assemblies in the 32P DSC, and account for 24-month operating cycles and the bounding 0.4

MTU assembly uranium loading. Therefore, there was no change from the method of evaluation described in the USAR.

STRUCTURAL EVALUATION

The weight calculations for the NUHOMS-32P DSC are described in Reference Series 2. These calculations use the same methodology as the NUHOMS-24P DSC weight calculations.

The following table provides a comparison of the methods used to analyze the DSC, Transfer Cask, and HSM.

DSC Basket

	CCNPP ISFSI USAR (24P) Method	32P Method
Analytical Code	ANSYS 5.2, 5.3	ANSYS 5.6, & 6.0
Dimensions / geometry	2D, 3D	3D
Materials behavior	Elastic/Plastic	Elastic/Plastic

The methodologies for the 32P are approved by their use in the USAR, except for one element, the ANSYS revision used. The changes from ANSYS 5.2 to 5.3 include features that are not used in the DSC basket analysis (see ANSYS release notes ANSYS 5.3.000663, June 1996), so ANSYS 5.2 and 5.3 may be regarded as identical for the purpose here. DCALC No. CA06630 benchmarks revisions 5.3, 5.6, and 6.0 to demonstrate that the results are essentially the same. Therefore, the change in an element of the methodology from ANSYS 5.2 and 5.3 to ANSYS 5.6 and 6.0 is not a departure from an approved method, in accordance with NEI 96.07 B4.3.8.1.

The exclusive use of 3D modeling for the 32P basket structural analysis is dictated by the tube and transition rail construction of the 32P basket, which differs from the 24P's tube and disc design. The structural analysis of the 32P basket is virtually identical to that of the TN-68 basket, a similar basket design approved by NRC CoC 72-1027.

DSC Shell

	CCNPP ISFSI USAR (24P) Method	32P Method
Analytical Code	ANSYS 5.2	ANSYS 5.6, 6.0, & 8.1
Dimensions / geometry	2D	2D
Materials behavior	Elastic	Elastic/Plastic, Elastic

The change from elastic to elastic/plastic materials behavior is an 'element of the methodology' change. Although elastic-plastic materials behavior was used for the 24P basket analysis, its approval by the NRC for a plate structure cannot be used to infer NRC approval of elastic/plastic analysis to a shell structure. The application of plastic analysis to a shell structure for the 32P is an element of a method that results in less conservative results as compared to elastic analysis to the plate structure used for the 24P. Hence, the change results in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses. This analysis was submitted to the NRC for approval with the response to the Request for Additional Information.

Transfer Cask

A combination of hand and finite element calculations were used for the transfer cask structural evaluation. The hand calculations are the same as the USAR methods. The finite element methods are as follows:

	CCNPP ISFSI USAR (24P) Method	32P Method
Analytical Code	ANSYS 5.2	See note
Dimensions / geometry	3D	See note
Materials behavior	Elastic	See note

Note: The geometry of the transfer cask is not changed. The only change is the increase of the DSC weight from the 24P to the 32P. The stress results of the USAR calculation are scaled up to include the effect of the increased 32P DSC weight.

HSM

The following HSM component structural capabilities are re-evaluated for the 32P DSC:

1. DSC support and miscellaneous steel: The results of the 24P analysis are scaled for the additional 32P DSC weight.
2. Horizontal storage module transfer cask restraint evaluation: The loading and resulting stresses of the 24P restraint evaluation are scaled for the increased weight of the 32P DSC.
3. DSC Seismic restraint calculation: The same hand calculation method that is used for the 24P is used to evaluate the 32P.
4. HSM concrete design: A combination of hand and finite element calculations were used for the HSM concrete structural evaluation. Following table lists the comparison of the HSM evaluations between the 24P DSC and 32P DSC.

	CCNPP ISFSI USAR HSM Evaluation (24P DSC)	HSM Evaluation (32P DSC)
Material Properties	No changes between 24P DSC and 32P DSC	
Soil Spring Evaluation	Spring constant is adjusted slightly for the 32P increased weight	
Model Representation	No changes between 24P DSC and 32P DSC	
Dead Load	The load is adjusted to account for the 32P increased weight	
Live Load	No changes between 24P DSC and 32P DSC	
Seismic Load	The seismic forces used for 32P evaluation are adjusted to account for the increased weight	
Thermal Load	Finite element models are used for thermal loads calculations. The model geometry, node and element numbers, cross-section properties, boundary conditions and material properties are identical for 24P and 32P evaluation. The only difference is 24P uses STRUDEL and 32P uses ANSYS. Since only beam elements are used in the analyses, the theory and all other inputs are identical, therefore there is no significant change in the analyses.	

In conclusion, two cases where a methodology departs from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses have been identified. One case involves crediting fixed and dissolved neutron absorber materials in lieu of burnup for criticality calculations. The other case involves an analysis change from elastic to an elastic/plastic analysis for all accident conditions except 100 psig applied to the inner pressure boundary. Prior NRC approval is required. The change in burnup versus fixed and dissolved neutron absorber credit was submitted to the NRC as part of the license amendment application. The second change involving elastic versus elastic/plastic material properties was sent to the NRC in follow-on correspondence related to the request for additional information for the 32P application.

References:

1. Reference Series 1 (Criticality Analysis)

   CCNPP Calculation No. CA05895, “The Criticality Benchmarks,” Revision 0

   CCNPP Calculation No. CA05896, “Criticality Analysis for Fuel Misloads and Accidents,” Revision 0

   CCNPP Calculation No. CA06227, “Criticality Analysis of the NUHOMS-32P for Calvert Cliffs ISFSI,” Revision 0

   CCNPP Calculation No. CA05673, “Criticality Analysis for a Dropped Fuel storage Cask, Revision 0

2. Reference Series 2 (DSC Structural Analysis)

   CCNPP Calculation No. CA06319, “Weight Calculations of DSC/TC System,” Revision 1

   CCNPP Calculation No. CA06335, "Basket Stress Analysis Due to Accident Transfer Drop Loads,” Revision 0

   CCNPP Calculation No. CA06325, "NUHOMS-32P Basket Rail Buckling Analysis - Accident,” Revision 0

   CCNPP Calculation No. CA06326, "NUHOMS 32P Basket Stress Analysis for Storage Loads (Normal and Accident),” Revision 0

   CCNPP Calculation No. CA06328, "NUHOMS-32P Dynamic Stress Strain Lead Properties at Different Temperatures," Revision 0

   CCNPP Calculation No. CA06331, "NUHOMS-32P DSC Lifting Fixture Stress Analysis," Revision 0

   CCNPP Calculation No. CA06332, "NUHOMS-32P DSC Bottom Stress Analysis for Lifting Loads," Revision 0

   CCNPP Calculation No. CA06334, "NUHOMS-32P Basket Buckling Analysis,” Revision 1
   CCNPP Calculation No. CA06336, "Basket Stress Analysis (Normal) for Transfer Loads,” Revision 0

   CCNPP Calculation No. CA06359 , “NUHOMS 32P - DSC Structural Analysis,” Revision 5

   CCNPP Calculation No. CA04678, “BGE (Calvert Cliffs Units 1 & 2) ISFSI Dry Storage Cask Drop Analysis”

   CCNPP Calculation No. CA04680, “ISFSI CE 14x14 Fuel Grid Horizontal Drop Evaluation”

   CCNPP Calculation No. CA05797, “DSC Horizontal Drop – Fuel Rod Cladding Integrity During Impact with a Broken Spacer Grid Fragment”

   CCNPP Calculation No. CA06630, Sensitivity study of ANSYS versions used in 32P structural analysis

3. Reference Series 3 (DSC Thermal Analysis)

   CCNPP Calculation No. CA06295, “Effective Fuel Properties,” Revision 0

   CCNPP Calculation No. CA06296, “Finite Element Model, Thermal Analysis,” Revision 0

CCNPP Calculation No. CA06297, "Transfer Thermal Analysis, 103°F Ambient," Revision 1

CCNPP Calculation No. CA06298, "Transfer Thermal Analysis, -3°F Ambient," Revision 0

CCNPP Calculation No. CA06300, "Maximum Operating Pressure, Storage and Transfer," Revision 1

CCNPP Calculation No. CA06305, "DSC Thermal Analysis – Normal Storage Conditions," Revision 0

CCNPP Calculation No. CA06306, "DSC Thermal Analysis - Off Normal Conditions (Max. Summer Temperature)," Revision 0

CCNPP Calculation No. CA06307, "DSC Thermal Analysis - Off Normal Conditions (Min. Winter Temperature)," Revision 0

CCNPP Calculation No. CA06308, "Effective Fuel Properties for Vacuum Drying," Revision 0

CCNPP Calculation No. CA06311, "Temperature Profile of the DSC," Revision 0

CCNPP Calculation No. CA06312, "Thermal Analysis of Storage Cases Poison Plates in Basket," Revision 0

CCNPP Calculation No. CA06314, "Thermal Analysis of Vacuum Drying," Revision 0

CCNPP Calculation No. CA06340, "Thermal Expansion Calculation for Transfer and Storage Conditions," Revision 0

CCNPP Calculation No. CA05760, "Effective Conductivity of the Reconfigured Fuel Assembly"

CCNPP Calculation No. CA06619, Rev. 0, Heat Transfer Analysis Comparison between ANSYS Versions 5.6, 5.7, and 6.0.

CCNPP Calculation No. CA06637, Rev. 0, Sensitivity Analysis of Homogenized Fuel Region

CCNPP Calculation No. CA06638, Rev. 0, Sensitivity Study of Structural Analysis due to the Mesh Density Changes of the Thermal Model

CCNPP Calculation No. CA06639, Rev. 0, Benchmarking Heat Transfer Analysis for ANSYS Versions 5.7 & 8.1

4. Reference Series 4 (Radiation Dose Analysis)

CCNPP Calculation No. CA06291, "Atomic Fractions for the Shielding Analysis of the NUHOMS 32P Basket," Revision 0

CCNPP Calculation No. CA06292, "NUHOMS-32P Radiation Dose Rates Loading and Transfer," Revision 0

CCNPP Calculation No.CA06293, "NUHOMS-32P – HSM Dose Rates for Calvert Cliffs ISFSI," Revision 0

CCNPP Calculation No. CA06327, "Shielding Evaluation with the New Top Shield Plug for NUHOMS-32P," Revision 0

Engineering Evaluation ES200200585, Evaluation of the Shielding Source Terms for the ISFSI 32P Phase I Design," Revision 0

CCNPP Calculation No. CA05924, "Calvert Cliffs ISFSI/NUHOMS-24P Radiation Dose Rates for Cask Loading and Transfer," Revision 1

CCNPP Calculation No. CA05925, Calvert Cliffs ISFSI/NUHOMS-24P HSM Dose Rates," Revision 0

CCNPP Calculation No. CA03902, "Fission Gas Release Dose Rates," Revision 0

5. Reference Series 5 (TC – Thermal Analysis)

   CCNPP Calculation No. CA06309, "Sensitivity of the Component Temperatures to the Position of the DSC in the Transfer Cask," Revision 0

   CCNPP Calculation No. CA06310, "Sensitivity of the Transfer Cask Thermal Analysis to the Axial Gaps Between DSC and Transfer Cask," Revision 0

   CCNPP Calculation No. CA06313, "Thermal Analysis of Transfer Cask with Poison Material in Basket," Revision 0

6. Reference Series 6 (HSM – Thermal Analysis)

   CCNPP Calculation No. CA06299, "Exit Air Temperature and Bulk Air Temperatures Within the HSM," Revision 0

   CCNPP Calculation No. CA06301, "HSM Thermal Analysis - Normal Storage Conditions," Revision 0

   CCNPP Calculation No. CA06302, "HSM Thermal Analysis – Max. Summer Temperature," Revision 0

   CCNPP Calculation No. CA06303, "HSM Thermal Analysis - Minimum Winter Temperature," Revision 0

   CCNPP Calculation No. CA06304, "HSM Thermal Analysis - Accident Conditions (Blocked Vents)," Revision 0

   CCNPP Calculation No. CA06341, "Maximum Temperature Gradient Across the Concrete Walls of the HSM," Revision 0

   CCNPP Calculation No. CA03945, Rev. 1, ISFSI - Revised Forrest Fire Evaluation.

   CCNPP Calculation No. CA6629,Rev. 0, "Forrest Fire Radiological Evaluation for the NUHOMS-32P HSM

   CCNPP Calculation No. CA06618, "Thermal Accident Blocked Vents for the DSC End Assemblies," Revision 0

7. Reference Series 7 (Structural Analysis)

   CCNPP Calculation No. CA04000, "NUHOMS 32P ISFSI Yard Paving and Approach Slab Evaluation," Revision 2

   CCNPP Calculation No. CA06329, "NUHOMS-32P – Transfer Cask Structural Analysis," Revision 2

   CCNPP Calculation No. CA06333, "NUHOMS-32P Lift Beam System Stress Analysis," Revision 0

   CCNPP Calculation No. CA06393, "NUHOMS 32P – CCNP Fuel Pool Floor and Cask Support Platform Evaluation," Revision 1

CCNPP Calculation No. CA06364, "NUHOMS 32P - CCNP ISFS HSM Facility Evaluation," Revision 0

CCNPP Calculation No. C-92-126, "ISFSI – ISFSF Access Road Pavement Design Evaluation," Revision 0

CCNPP Calculation No. C-92-127, "ISFSI – ISFSF Access Road Utility Tunnel Evaluation," Revision 0

CCNPP Calculation No. C-92-128, "ISFSI – ISFSF Access Road Pavement Design Evaluation," Revision 0

CCNPP Calculation No. C-92-129, "Evaluation of Aux. Building Slab at EL. 45' for ISFSI Cask Transport Vehicle Loads. (NUHOMS 24P and 32P), Revision 1

CCNPP Calculation No. C-92-130, "NUHOMS-32P Transfer Cask Drop on Safety Related Piping and Ductbank," Revision 1

CCNPP Calculation No. C-92-267, "Overturning & Sliding Evaluation for NUHOMS 32P ISFSI Onsite Transfer Cask," Revision 0

CCNPP Calculation No. C-93-151, "NUHOMS-32P Transfer Cask Curb Drop Evaluation," Revision 2

CCNPP Calculation No. CA04679, "ISFSI Fuel Assembly Upper End Fitting Structural Integrity"

CCNPP Calculation No. CA06499, "32P DSC Loading and Unloading Loads," Revision 2

8. Calvert Cliffs Independent Spent Fuel Storage Installation USAR, Revision 12
9. Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type), American National Standard, ANSI/ANS 57.9-1984
10. Constellation Energy Group letter to the U.S. Nuclear Regulatory Commission, "Revision to the Technical Specifications to Support the ISFSI NUHOMS-32P Upgrade," Dated December 12, 2003
11. Ichikawa, T and Koizumi, H, "Possibility of Radiation-induced Degradation of Concrete by Alkali-Silica Reaction of Aggregates," Journal of Nuclear Science and Technology, Vol. 39, P. 880-884, August 2002
12. "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel NUHOMS®-24P," NUH-002, Revision 1a, July-1989
13. NRC - Safety Evaluation Report for the Baltimore Gas and Electric Company's Safety Analysis Report for an Independent Spent Fuel Storage Installation at Calvert Cliffs, November 1992
14. Transnuclear E-20112, "Procurement Specification for Borated Aluminum Sheets for the NUHOMS-32P Dry Shielded Canister," Revision 3
15. Calvert Cliffs Nuclear Power Plant Technical Procedure ISFSI-02, Independent Spent Fuel Storage Installation (ISFSI) Unloading, Revision 8
16. Nuclear Regulatory Commission Safety Evaluation Report, Docket No. 72-1004, Standardized NUHOMS Modular Storage System for Irradiated Nuclear Fuel, Certificate of Compliance No. 1004, Amendment 5

17. CCNPP Specification SP-0564C, "NUHOMS-32P Dry Shielded Canister Design Specification"
18. PNL-6189, Levy, I.S., et al., Pacific Northwest Laboratories, "Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircaloy Clad Fuel Rods in Inert Gas" May 1987.
19. Safety Evaluation Supporting Issuance of License for Baltimore Gas & Electric Co. ISFSI, SNM-2505
20. CA04680,Rev. 0 (HABGE-02/98-610, R4), (24P) DSC Structural Analysis

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: The proposed activity is the use of a new design of Dry Shielded Canister (DSC), NUHOMS-32P, for storing spent fuel at the Calvert Cliffs Nuclear Power Plant's (CCNPP's) Independent Spent Fuel Storage Installation (ISFSI). The use of NUHOMS-32P DSCs will be made in addition to the use of NUHOMS-24P DSCs. These two types of DSCs have the same external dimensions; however, the NUHOMS-32P DSC can store 32 fuel assemblies. NUHOMS-32P Basket design is based on Transnuclear TN-68 design, which is approved by the NRC.

There are no physical changes being made to the Transfer Cask (TC) or the Horizontal Storage Modules (HSMs). All the major steps for loading a DSC (vacuum drying, welding, etc.) are the same for the NUHOMS-24P DSC and the NUHOMS-32P DSC systems.

The activity also consists of the following additional fuel assembly changes, which are not directly associated with the new NUHOMS-32P DSC design, but have been included in the design analysis of NUHOMS-32P DSC.

- Increase of fuel assembly weight from 1,300 lbs. to 1,450 lbs. This accounts for the weight increase during irradiation and the possible storage of control components with the assembly.
- A limiting assembly mass of 0.400 MTU, compared to the previously used nominal value of 0.386 MTU. This bounds all standard CE 14x14 fuel assemblies used at CCNPP.
- Increase in fuel assembly neutron source design basis in calculating the radiation dose, in order to account for higher burnups in lower enriched assemblies.

Reason for Activity: This proposed activity will increase the ISFSI storage capacity. The NUHOMS-32P DSC design also allows CCNPP to reduce the minimum number of canister loadings each year from four (using the NUHOMS-24P DSC design) to three (with the NUHOMS-32P DSC design). This should reduce the total annual radiological dose associated with the canister loading.

Activity Evaluation: New analyses are performed to verify that confinement, shielding, criticality control, structural stresses, and passive heat removal are acceptable with the use of NUHOMS-32P DSCs at the CCNPP ISFSI.

The results of the analyses demonstrate that the new DSC and the existing TC and HSM with the use of the new DSCs meet all of the design criteria, and will provide for a safe storage of the spent fuel assemblies under normal, off-normal and postulated accident conditions.

Conclusion: The proposed activity has been evaluated against the eight criteria of 10CFR 72.48(c)(2). Criteria 7 and 8, which concern the design basis limits for the fission product barrier and a departure from a method of evaluation described in the USAR, are not met. It is concluded that the proposed activity will require a License Amendment prior to its implementation.

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Revision 1

ATTACHMENT 3, SAFETY EVALUATION FORM

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ACTIVITY: Calvert Cliffs ISFSI USAR Change **50.59 Log No.**____ **or 72.48 Log No.** 94-0-101-001

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

___YES X NO Involve an Unreviewed Safety Question (USQ)?
___YES X NO Involve a change to the Technical Specifications/License Conditions or Bases?
X YES ___ NO Require a change or addition to the UFSAR or USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

___YES X NO Involve a Significant Increase in Occupational Dose?
___YES X NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Sam Shakir SAM SHAKIR Department: CCSO Date: 8/31/94
PRINTED NAME AND SIGNATURE (VECTRA)

X YES ___ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: Robert H. Beall PRINTED NAME	Resp. Ind.: JOHN B. MAKAR PRINTED NAME	Resp. Ind.: Patricia A. Jones PRINTED NAME
R H Beall SIGNATURE	OSS for J.B. MAKAR Per Telecon. SIGNATURE	SIGNATURE
Work Group: Fuel Management	Work Group: System Engineer	Work Group: Radiation Safety
Date: 8/31/94	Date: 8/31/94	Date: 8/31/94

Approved ✓ Disapproved ___
Signature OSS for M-TAYLOR Per Telecon
INDEPENDENT REVIEWER (VECTRA)
Date 8/31/94

Approved ✓ Disapproved ___
Signature Michael J. Gahan 9.8.94
GS-DES GS-TSEB, or PE-PDSU
OSS for J.H.RILEY, CCSO 8/31/94 Per Telecon.
Date

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 94-139 Date: 8-12-94

Recommend Approval ✓ Recommend Disapproval___ Signature___ Date: 8-12-94
POSRC CHAIRMAN

Approved ✓ Disapproved___ Signature___ Date: 9/12/94
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
OSSRC Meeting No.: 95-003 Date:___

Recommend Approval___ Recommend Disapproval___ Signature___ Date:___
OSSRC CHAIRMAN

WORK GROUP
Licensing Unit
Reviewed By: Getachew Tesfaye
Getachew Tesfaye 8-31-94

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: Calvert Cliffs ISFSI USAR Change **50.59 Log No._____ or 72.48 Log No.** 94-0-101-001

Proposed Activity:
To allow closure welds on the DSC shield plug and top cover plate to be made manually in addition to the welding made by the automated welding machine. Manual welding is already allowed for sealing the vent ports on the DSC. That task is listed in Table 7.4-1 of the ISFSI USAR as Seal Weld Penetration Plug. Manual welding for closure welds shall be included within that task. This will result in the following changes to the ISFSI USAR:
1) Change Volume I, Section 1.3.1.8. to read:
"The DSC closure welds on the shield plug and the top cover plate are normally placed by a fully remote, automatic welding system. The system includes modular ... to remove the shield plug and top cover plate closure welds. Manual welding may be used for making closure welds and to substitute for automatic welding when the automatic welding equipment is temporarily unavailable. The allowed duration of manual welding is limited by the ambient dose rate at the location of the welding."
2) Change the description of the seal weld penetration plug task in Table 7.4-1 to read:
"Seal Weld Penetration Plug and Other Manual Welding."
The appropriate ISFSI procedure will be revised to add manual welding in accordance with the ISFSI USAR change.
Reason for Activity:
Manual welding is more efficient than automatic welding in some cases for making closure welds. Manual welding also allows the continuation or completion of welding operations when the automatic welding equipment is temporarily unavailable.

Function (s) of affected SSC:
The only SSC affected by the welding method of the top shield plug and top cover plate to the Dry Shielded Canister (DSC) is the DSC itself. The DSC provides containment and confinement of the spent fuel during storage. The closure welds are part of the containment and confinement boundary.

ISFSI USAR Sections Reviewed:
Vol. I, Section 1.3.1.8, Vol. I, Section 3.3.2.1, Vol. I, Section 5.1.1.3, Vol. I, Section 5.1.1.4, Vol. I, Table 7.4-1, Vol. I, Section 8.2, Vol. I, Section 10.3.2.3, Vol. I, Section 10.3.2.4, Vol. IV, NRC ISFSI SER, Section 2.2.4.4, Vol. V, Technical Specification 3/4.2, Vol. V, Technical Specification Basis 3/4.2.

Complete for 50.59 and 72.48:
1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

____Yes _X_ No May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

The function of the Dry Shielded Canister as a containment and confinement barrier is not affected by the welding method (manual or automatic) for the closure since the manual welds are made in accordance with the requirements of the Welding Procedure Specification WPS P8-T or P8-T-LH (Manual) and must be nondestructively tested. This procedure is equivalent to WPS P8-T (Machine) used for the automatic machine welding. This procedure and the nondestructive testing will assure the quality and integrity of the welds. Therefore, the probability of a malfunction is not increased by this change.

____Yes _X_ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Any welding placed manually will be made to the same specification and must pass the same testing requirements as that made by the automatic welder. Therefore, this activity does not increase the

ACTIVITY: <u>Calvert Cliffs ISFSI USAR Change</u> **50.59 Log No.____ or 72.48 Log No.** <u>94-0-101-001</u>

consequences of a weld malfunction. The occupational dose consequences for the use of manual welding in place of the automatic remotely operated welder are addressed in the answer to question 3.

____ Yes __X__ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Two accident scenarios, a drop accident and a leakage accident, are addressed in the ISFSI USAR that consider a breach in the containment and confinement boundary formed by the canister closure welds. The probability of these accident is not increased by the proposed change since the integrity and quality of the manual welds will be as good as those made by the automatic welder. The manual welds performed by qualified welders will be placed in accordance with the requirements of WPS P8-T or P8-T-LH (manual), and must pass nondestructive testing. Therefore, the welding method (manual or automatic) is not relevant to the probability of an accident since both welding methods are subject to the same quality and integrity requirements.

____Yes __X__ No May the consequences of an accident previously evaluated in the SAR be increased?

The consequences of a drop accident causing failure in the canister closure welds, or the consequences of a DSC leakage accident due to a weld leak are not affected by the welding method.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not increased.

____Yes __X__ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

No new malfunctions can be caused by the canister closure welding method since the closure welds are done in accordance with all applicable codes, standards and procedures, and must pass the nondestructive testing.

____Yes __X__ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

No new accidents can be caused by the canister closure welding method since the closure welds are done in accordance with all applicable codes, standards and procedures, and must pass the nondestructive testing .

<u>Complete for 50.59 and 72.48:</u>

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

____Yes __X__ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

ATTACHMENT 3, SAFETY EVALUATION FORM

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ACTIVITY: Calvert Cliffs ISFSI USAR Change__ 50.59 Log No.____ or 72.48 Log No. 94-0-101-001

Bases	Discussion of why the margin of safety is not reduced
3/4.2	Section 3/4.2 states that the safety analysis of leak tightness of the DSC is based on a weld being leak tight to 10^{-4} atm-cc/s. The proposed change does not change the leak rate criteria. The margin of safety is therefore not reduced.

Complete for 72.48:

___Yes _X_ No Will the proposed activity involve a significant increase in occupational dose?

The estimated personnel dose for all manual welding including the seal weld penetration plug task will remain unchanged at 65.3 mrem, as shown in Table 7.4-1 of the ISFSI USAR. The number of people does not have to be increased to prevent an individual from exceeding any limit of 10 CFR 20. Difficult weld geometry's are encountered when making closure welds, particularly in the keyway area and in weld repairs, requiring multiple setups of the automatic welding machine. Manual welding could replace some of the time needed to manually reset the automatic welder on top of the DSC. The field could then use that time to complete the weld manually instead of resetting the automatic welder several times to do that task. This results in a more efficient operation without increasing the personnel collective dose.

___Yes _X_ No Will the proposed activity involve a significant unreviewed environmental impact?

The welding method (manual or automatic) for the canister closure welds does not affect any area of the plant site previously undisturbed for the ISFSI or require a revision to the ISFSI Environmental Impact Statement.

Summary: (For NRC Report, provide a brief overview)

The ISFSI USAR (Vol. I, Section 1.3.1.8) describes the Dry Shielded Canister weld closure on the shield plug and top cover plate as being performed by a fully remote, automatic welding system. This description is changed to allow manual welding for making closure welds and to substitute for the automatic welding equipment when it is temporarily unavailable. Manual welding can safely and efficiently replace the remote welding system for making closure welds, since resetting the automatic welding system is a more complex effort that results in similar occupational exposure to that obtained from performing the closure welds manually. The allowed duration of manual welding is limited by the ambient dose rates at the location of the welding. This will ensure that the personnel dose for the task does not significantly exceed the estimated dose in table 7.4-1 of the ISFSI USAR. This change does not constitute an unreviewed safety question, a significant increase in occupational exposure or an unreviewed environmental impact for the Independent Spent Fuel Storage Installation.

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 1

This screening is for: ______**10 CFR 50.59 Applicability** __x__ **10 CFR 72.48 Applicability**
(Check one regulation only)

______**CCNPP** __x__**ISFSI**
(Check one facility only)

(Check one activity type only)

______**Procedure:** Procedure No./Change No.:__________

______**Temporary Alteration:** Temporary Alteration No.:__________

______**Setpoint Change:** SCAF No(s):__________

______**Modification:** MCR/FCR/FEC No.:__________
FEC Supplement No.:__________

______

______**Core Reload:** Unit and Cycle:__________

__x__**UFSAR/USAR:** UFSAR/USAR Change No.: 94-29

______**Other:** Identify Activity Type:__________

Brief description of the activity:

To allow closure welds on the DSC shield plug and top cover plate to be made manually in addition to the automatic welding system. The manual welding is more efficient in some cases of closure welds and it could allow continuation or completion of welding operations when the automatic welding equipment is temporarily unavailable. This activity will involve a change to the ISFSI USAR Vol. I, Section 1.3.1.8, and Table 7.4-1.

Technical Specifications/License Conditions (10 CFR 50.59/72.48)

1.____YES __x__NO Is the proposed activity a change or will it cause a change to the Technical Specifications/License Conditions or Bases?

2.____YES __x__NO Will the proposed activity cause Structures, Systems or Components (SSCs) to be operated in a manner that violates the Technical Specifications/License Conditions or Bases?

If both answers are "No," continue with the screening. Justification for each "No" answer shall be provided. List the sections of the Technical Specifications/License Conditions that were reviewed.

Justification:

The change to the ISFSI USAR description of the automated closure welding operation of the Dry Shielded Canister to allow closure welds to be made manually instead of using the automatic remote welding system does not impact any technical specification. All final welds will meet the original ISFSI Tech. Spec. requirements.

Technical Specifications/License Condition Sections Reviewed:

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 2

Reviewed all sections of the ISFSI Technical Specification manual.

If either of the above answers is "Yes," complete a Safety Evaluation and consult CCI-143 for License Amendment Proposals.

CCNPP/ISFSI Facility (10 CFR 50.59/72.48)

1.__x__YES _____NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of the structure, system or component (SSC) directly affected by the activity?

If "No," answer each question below:

Why is the SAR description of the function of the SSC not affected?

Why is the SAR description of the method of performing the function of the SSC not affected?

Why is the SAR description of the design of the SSC not changed?

2.____YES __x__NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of any other SSC described in the SAR?

If "No," answer the following question:

Explain why the activity does not affect other SSCs described in the SAR..

The activity will allow the use of manual welding, in addition to the automatic welding system, for closure welds during the closure operation of the DSC. The manual weld will be made in accordance with the Welding Procedure Specification WPS P8-T or P8-T-LH (Manual) and must be nondestructively tested. The quality and integrity of the manual weld is as good as the weld placed by the automatic welder. This activity will not affect other SSCs described in the ISFSI USAR.

3.__x__YES ______NO Is the proposed activity a revision to the SAR. (Editorial changes are limited to obvious grammatical/spelling errors, reorganization of portions of the SAR or minor changes that do not affect the intent of the information conveyed by a drawing.)

4.__x__YES ______NO Will the proposed activity add to or delete from the SAR description of a SSC?

Procedures (10 CFR 50.59/72.48)

1.__x__YES _____NO Will the proposed activity affect the intent of any procedure described in the SAR (editorial changes do not need a Safety Evaluation)? The NRC staff does not consider procedures simply listed in the SAR to be described in the SAR. Also, procedures include anything that defines or describes activities or controls over functions, tasks, reviews, tests and safety review meetings.

2._____YES __x__NO Will the proposed activity cause SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: This activity will change the appropriate ISFSI procedures to allow for closure welds to be made manually in addition to using the automatic welding system. The manual weld will be made in accordance with the requirements of the Welding Procedure Specification WPS P8-T or P8-T-LH (Manual). The manual weld shall be of the same characteristics as the weld placed by the automatic welder. Therefore, manual welding shall not impact the design, function, or method of performing the function of the DSC, the top cover plate, or shield plug.

Tests or Experiments (10 CFR 50.59/72.48)

1._____YES __x__NO Will the proposed activity result in conducting a test or experiment causing SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: This activity is not a test or experiment.

ISFSI (10 CFR 72.48) These questions are only required to be answered for activities affecting ISFSI.

1.__x__YES ______NO Will the proposed activity increase any occupational dose for ISFSI related activities?

2._____YES __x__NO Will the proposed activity use additional property for ISFSI operations?

3._____YES __x__NO Will the proposed activity add or change the roads or transport equipment, including cranes, used for ISFSI operations?

Justify each "No" answer below:

Justification: This activity allows manual welding for closure welds on the DSC top cover plate and shield plug which is performed in the Cask Wash Pit on the 69' level of the Auxiliary Building. No additional ISFSI property, changes to the road, or transport equipment is required or included in this activity.

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 4

SAR Sections Reviewed:

Volumes I, IV, & V of the ISFSI USAR

If ALL answers are "No", A Safety Evaluation is not required.

If ANY answer is "Yes", A Safety Evaluation is required.

1.___x__YES ______NO Does this activity require additional screening?

10CFR 50.59 For Impact on CCNPP
10 CFR 72.48 For Impact on ISFSI

If "Yes", Perform a separate Safety Evaluation Screening.

Prepared By: Sam Shakir SAM SHAKIR Date: 8/30/94
PRINTED NAME AND SIGNATURE

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 1

This screening is for: __x__10 CFR 50.59 Applicability _____ 10 CFR 72.48 Applicability
(Check one regulation only)

__x__CCNPP _____ISFSI
(Check one facility only)

(Check one activity type only)

______**Procedure:** Procedure No./Change No.:__________________

______**Temporary Alteration:** Temporary Alteration No.:__________________

______**Setpoint Change:** SCAF No(s):__________________

______**Modification:** MCR/FCR/FEC No.:__________________
FEC Supplement No.:__________________

______

______**Core Reload:** Unit and Cycle:__________________

__x__**UFSAR/USAR:** UFSAR/USAR Change No.: 94-29

______**Other:** Identify Activity Type:__________________

Brief description of the activity:

To allow closure welds on the DSC shield plug and top cover plate to be made manually in addition to the automatic welding system. The manual welding is more efficient in some cases of closure welds and it could allow continuation or completion of welding operations when the automatic welding equipment is temporarily unavailable. The welding operation takes place inside the Auxiliary Building.

Technical Specifications/License Conditions (10 CFR 50.59/72.48)

1._____YES __x__NO Is the proposed activity a change or will it cause a change to the Technical Specifications/License Conditions or Bases?

2._____YES __x__NO Will the proposed activity cause Structures, Systems or Components (SSCs) to be operated in a manner that violates the Technical Specifications/License Conditions or Bases?

If both answers are "No," continue with the screening. Justification for each "No" answer shall be provided. List the sections of the Technical Specifications/License Conditions that were reviewed.

Justification:

The description of the automated closure welding operation appears only in the ISFSI USAR and the ISFSI Tech. Spec. No such description appears in the UFSAR or the plant Technical Specification. Therefore, allowing closure welds to be done manually in addition to using the automatic remote welding system is strictly an ISFSI change and does not impact the plant Tech. Spec. All final welds will meet the original ISFSI Tech. Spec. requirements.

Technical Specifications/License Condition Sections Reviewed:

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 2

Reviewed all sections of the CCNPP Technical Specification, none are applicable to this activity.

If either of the above answers is **"Yes,"** complete a Safety Evaluation and consult CCI-143 for License Amendment Proposals.

CCNPP/ISFSI Facility (10 CFR 50.59/72.48)

1._____YES ___x__NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of the structure, system or component (SSC) directly affected by the activity?

If "No," answer each question below:

Why is the SAR description of the function of the SSC not affected?

The activity has no impact on the function of the welded components (DSC, shield plug, and top plate). All these components are part of the ISFSI and are described in the ISFSI USAR. No SSCs described in the UFSAR are affected by this activity. Therefore, this activity does not affect the function of any SSCs in the Auxiliary Building.

.

Why is the SAR description of the method of performing the function of the SSC not affected?

This activity affects the welding closure operation of the DSC. This operation is only described in the ISFSI USAR but not in the UFSAR. Therefore, allowing some closure welding to be performed manually instead of using the automatic welding system has no impact on the method of performing the function of any SSCs in the Auxiliary Building.

Why is the SAR description of the design of the SSC not changed?

Allowing manual welding in the DSC closure operation is convenient and efficient. It does not affect the design of the DSC, which is an ISFSI component. No other SSCs in the Auxiliary Building, where the welding operation takes place, are affected by this activity.

2._____YES ___x___NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of any other SSC described in the SAR?

If "No," answer the following question:

Explain why the activity does not affect other SSCs described in the SAR..

The activity will allow the use of manual welding, instead of the automatic welding system, for making welds during the closure operation of the DSC. The manual weld will be made in accordance with the Welding Procedure Specification WPS P8-T or P8-TLH (Manual) and must be nondestructively tested. The manual weld will be as good as the weld made by the automatic welder. No other SSCs are affected by this activity.

3._____YES ____x____NO Is the proposed activity a revision to the SAR. (Editorial changes are limited to obvious grammatical/spelling errors, reorganization of portions of the SAR or minor changes that do not affect the intent of the information conveyed by a drawing.)

4. YES [✓ NO] Will the proposed activity add to or delete from the SAR description of a SSC?

Procedures (10 CFR 50.59/72.48)

1. YES [✓ NO] Will the proposed activity affect the intent of any procedure described in the SAR (editorial changes do not need a Safety Evaluation)? The NRC staff does not consider procedures simply listed in the SAR to be described in the SAR. Also, procedures include anything that defines or describes activities or controls over functions, tasks, reviews, tests and safety review meetings.

2. YES [✓ NO] Will the proposed activity cause SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: The activity allows for closure welds to be done manually instead of using the automatic welding system. This change does not affect any procedures outlined in the UFSAR. The welds made by manual welding shall be of the same characteristics as the weld placed by the automatic welder. Therefore, manual welding shall not impact the design, function, or method of performing the function of any SSCs described in the UFSAR and located in the Auxiliary Building where the welding operation takes place.

Tests or Experiments (10 CFR 50.59/72.48)

1. YES [✓ NO] Will the proposed activity result in conducting a test or experiment causing SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: This activity is not a test or experiment.

ISFSI (10 CFR 72.48) These questions are only required to be answered for activities affecting ISFSI.

1. YES NO Will the proposed activity increase any occupational dose for ISFSI related activities?

2. YES NO Will the proposed activity use additional property for ISFSI operations?

3. YES NO Will the proposed activity add or change the roads or transport equipment, including cranes, used for ISFSI operations?

Justify each "No" answer below:
Justification:

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 4

SAR Sections Reviewed:

Chapters 11 and 14 of the UFSAR. None are applicable to this activity.

If <u>ALL</u> answers are "No", A Safety Evaluation is not required.

If <u>ANY</u> answer is "Yes", A Safety Evaluation is required.

1. __x__ YES ______ NO Does this activity require additional screening?

10CFR 50.59 For Impact on CCNPP
10 CFR 72.48 For Impact on ISFSI

If "Yes", Perform a separate Safety Evaluation Screening.

Prepared By: Sam Shakir SAM SHAKIR Date: 8/30/94
PRINTED NAME AND SIGNATURE

CCI-177
Rev. B/Change 0

ATTACHMENT 2, UFSAR CHANGE REQUEST FORM (UCR)

NONMOD # 94-29

To: UFSAR Coordinator**Error! Bookmark not defined.**
From: Sam Shakir (Printed Name) Work Group CCSO Date 8/16/94
Phone Number: 2179 System Number 101

SECTION 1 (Change Initiation)
UFSAR CHANGE SOURCE DOCUMENT

FCR/FEC/MCR # ______ (Circle One) Safety Evaluation Log # 94-0-101-01

RDC ______ Procedure # ______

License Amendment # ______

Regulatory Generic Correspondence # ______ (Generic Letter, Bulletin or Information Notice)

Unit 1 ____ Unit 2 ____ Common ____ ISFSI X

DESCRIPTION OF UFSAR CHANGE:
1) Change Volume I, Section 1.3.1.8. to read:
"The DSC closure welds on the shield plug and the top cover plate are normally placed by a fully remote, automatic welding system. The system includes modular ... to remove the shield plug and top cover plate closure welds. Manual welding may be used for making closure welds and to substitute for automatic welding when the automatic welding equipment is temporarily unavailable. The allowed duration of manual welding is limited by the ambient dose rate at the location of the welding."

2) Change the description of the seal weld penetration plug task in Table 7.4-1 to read:
"Seal Weld Penetration Plug and Other Manual Welding" (see attached markup of table 7.4-1).

UFSAR SECTIONS AFFECTED: [Attach Marked up Page(s)]
Volume I, Section 1.3.1.8.
Table 7.4-1.

CCI-177
Rev. B/Change 0

ATTACHMENT 2, UFSAR CHANGE REQUEST FORM (UCR)

NON MOD # 94-29

SECTION 2 *(Interdisciplinary Reviews)*

RESP. IND. ______________________ WORK GROUP: ______________________
Printed Name and Signature

RESP. IND. ______________________ WORK GROUP: ______________________
Printed Name and Signature

RESP. IND. ______________________ WORK GROUP: ______________________
Printed Name and Signature

SECTION 3 *(Implementation Verification Prior to UFSAR Incorporation)*

VERIFICATION THAT PLANT MODIFICATION OR AS-BUILT INFORMATION HAS BEEN IMPLEMENTED:

☐ Partial Implementation

(For changes which have been partially implemented, identify the completed portion of the change on the marked-up UFSAR pages. If implementation is complete on one unit only, check the appropriate box, below.)

☐ Unit 1 ☐ Unit 2

RESPONSIBLE ENGINEER: ______________________ DATE: __________

SECTION 4 *(Final Review/Approval Prior to UFSAR Incorporation)*

FINAL REVIEW & APPROVAL OF THIS CHANGE:

RESPONSIBLE ENGINEER: Sam Shakir SAM SHAKIR DATE: 8/30/94

RESP. ENGR'S. SUPERVISOR: Ass for M. TAYLOR per Telecon DATE: 8/30/94

UFSAR COORDINATOR: ______________________ DATE: __________

PE-LICENSING UNIT OR WGL: ______________________ DATE: __________

Table 7.4-1

Estimated Occupational Exposure for One HSM Load

Page 1 of 2

[See Reference 7.11]

CHANGE

Operation	Number of Personnel	Effective Time in Radiation Field (hours)	Average Distance from Source (feet)	Ambient Dose Rate (mrem/hr)	Dose Per Worker (mrem)	Total Personnel Dose (mrem)
LOCATION: Fuel Pool						
Load Fuel into DSC	4	10.00	30.0	2.0	20.0	80.0
LOCATION: Cask Decon Pit						
Decontaminate Outer Surface of Cask	2	1.00	1.5	83.6	83.6	167.2
Decontaminate Shield Plug and Exposed DSC Shell	1	1.00	1.5	41.4	41.4	41.4
Lower Water Level in DSC Cavity	2	0.25	4.0	10.4	2.6	5.2
Set up Automatic Welder to Weld Lead Plug to DSC	2	0.25	1.5	41.4	10.4	20.7
Perform Dye Penetrant Examination	1	1.50	1.5	41.4	62.1	62.1
Remove Remaining Water and Vacuum Dry DSC Cavity	2	1.00	4.0	10.4	10.4	20.8
Drain Cask/DSC Annulus	2	0.25	1.5	83.6	20.9	41.8
Backfill DSC Cavity with Helium	2	0.25	4.0	75.3	18.8	37.7
Perform Helium Leak Test	1	0.50	1.5	130.6	65.3	65.3
Seal Weld Penetration Plug AND OTHER MANUAL WELDING	1	0.50	1.5	130.6	65.3	65.3
Perform Examination On Penetration Plug Welds	1	0.25	1.5	130.6	32.7	32.7
Install Top Cover Plate	2	0.25	1.5	66.7	16.7	33.4
Set Up Automatic Welder To Weld Top Cover Plate To DSC	2	0.25	1.5	66.7	16.7	33.4

of the DSC to the HSM. Both solid neutron and lead gamma shielding are incorporated into the transfer cask design. Figure 1.3-2 shows the major components of the transfer cask. The Calvert Cliffs transfer cask has a solid hydrogenous neutron shield in the outer annulus of the cask, and as a result the liquid neutron shield expansion tank of Reference 1.2 is deleted.

1.3.1.4 Transfer Trailer [See Reference 1.4]

The transfer trailer is used to transport the transfer cask skid and the loaded transfer cask from the Auxiliary Building to the ISFSI. The transfer trailer is an industrial heavy-haul trailer with pneumatic tires, hydraulic suspension and steering, and brakes on all wheels. Four hydraulic jacks are incorporated into the transfer trailer design to provide vertical elevation adjustment for alignment of the cask at the HSM. The transfer trailer is shown in Figure 1.3-3. It is pulled by a conventional tractor.

1.3.1.5 Transfer Cask Skid and Positioning System

The transfer cask skid is essentially identical in design and operation to previous NUHOMS-24P system transfer cask support skids. The skid is supported on lubricated bearing plates attached to the trailer deck and can be moved horizontally on the bearing plates by the hydraulic actuators of the skid positioning system. The skid is secured to the trailer deck in a travel lock position during cask loading and transport operations. The transfer cask skid is shown in Figure 1.3-4.

1.3.1.6 Hydraulic Ram System

The hydraulic ram consists of a double acting hydraulic cylinder with a capacity of 80,000 lb. in either push or pull and stroke of 21 feet. The ram will be supported during operation by a frame assembly attached to the bottom of the transfer cask and a tripod assembly resting on the concrete slab. The operational loads of the hydraulic ram are grounded through the transfer cask. The hydraulic ram system includes a grapple at the end of the piston which is used to engage a grapple ring on the DSC for retrieval operations. Figure 1.3-5 shows the hydraulic ram system.

1.3.1.7 Vacuum Drying System

The vacuum drying system removes water and air from the DSC and fills it with helium. The vacuum drying system has four operational modes: water removal, helium forced water removal, vacuum pumping, and helium backfilling.

1.3.1.8 Automated Closure Welding System

The DSC closure welds on the shield plug and the top cover plate are placed by a fully remote, automatic welding system. The system includes modular components and is designed for rapid setup. Welding operations are remotely controlled by an operator who views the progress of the weld through closed circuit television. The welding head is designed to permit rapid replacement with either a UT probe, or a plasma gouging torch which can be used to remove the shield plug and top cover plate closure welds.

CHANGE

ATTACHMENT 3, SAFETY EVALUATION FORM

Page 1 of 4

ACTIVITY: Storage of empty DSC's at Calvert Cliffs ISFSI **50.59 Log No.**____ **or 72.48 Log. No.** 94-0-101-002

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

___YES _X_ NO Involve an Unreviewed Safety Question (USQ)?
___YES _X_ NO Involve a change to the Technical Specifications/License Conditions or Bases?
_X_YES ___ NO Require a change or addition to the UFSAR or USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

___YES _X_ NO Involve a Significant Increase in Occupational Dose?
___YES _X_ NO Involve a Significant Unreviewed Environmental Impact?

Sam Shakir
Prepared by: SAM SHAKIR Department: CCSO Date: 8/2/94
PRINTED NAME AND SIGNATURE (VECTRA)

_X_ YES ___ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Testaye PRINTED NAME	Resp. Ind.: J.B. MAKAR PRINTED NAME	Resp. Ind.: Robert H. Beall PRINTED NAME
SIGNATURE	SIGNATURE	SIGNATURE
Work Group: Licensing	Work Group: System Engineer	Work Group: Fuels Management
Date: 8/10/94	Date: 8-05-94	Date: 8/10/94

Approved ✓ Disapproved ___	Approved ✓ Disapproved ___
Signature Sam Shakir for Moses Taylor INDEPENDENT REVIEWER (VECTRA) Per Tele-CON.	Signature Michael J. Gahan GS-DES ~~GS-TSES, or PE-PDSU~~ M.J. GAHAN JWR CCSO 8/10/94 9.20.94
Date 8/4/94	Date

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 94-145 Date: 9-28-94

Recommend Approval ✓ Recommend Disapproval____ Signature____ Date 9-28-94
POSRC CHAIRMAN

Approved ✓ Disapproved____ Signature____ Date 9/28/94
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
OSSRC Meeting No.: 95-003 Date:____

Recommend Approval____ Recommend Disapproval____ Signature____ Date____
OSSRC CHAIRMAN

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: Storage of Empty DSC's at Calvert Cliffs ISFSI **50.59 Log No.____ or 72.48 Log. No. 94-0-101-002**

Proposed Activity:
This activity evaluates the effects of using the Independent Spent Fuel Storage Installation (ISFSI) site for storage of new empty Dry Shielded Canisters (DSCs) horizontally on cribbing inside the security fence which surrounds that area. The DSCs are Stainless Steel cylindrical shells that when filled provide confinement of radioactive spent fuel. The DSCs and spent fuel are transferred from the Spent Fuel Pool and stored in the concrete Horizontal Storage Modules (HSMs) at the ISFSI site. The orientation of the stored empty DSCs will be such that their ends are in the north-south direction facing the HSMs. The empty DSCs will be stored at a distance away from the HSMs enough to allow for normal spent fuel transportation and storage activities. The activity will result in the following change to the ISFSI USAR to allow the storage of these empty DSCs:
Add the following to Volume I, Section 4.1.1: "The ISFSI site may be utilized for storage of empty DSCs. The empty DSCs may be stored there until they are needed for spent fuel loading and permanent storage. The empty DSCs will be stored horizontally on wood cribbing with their ends facing north-south at a distance from the HSMs to allow for normal spent fuel transportation and storage activities."

Reason for Activity:.
The 20 empty canisters available at Calvert Cliffs require storage until they can be used in the transfer and storage of spent fuel. The ISFSI site provides a convenient and secure laydown storage area for these empty canisters until they are utilized.

Function (s) of affected SSC:
The HSMs at the ISFSI house spent fuel in DSCs and provide physical protection for the canisters, radiation shielding and flow paths for natural circulation heat dissipation.
SAR Sections Reviewed: ISFSI SAR Vol. I, Sections 1.2.1, 4.1.1, 8.2.2.2 and 8.2.7.

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

____Yes _X_ No May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Storage of empty DSCs at the ISFSI site does not affect the function of the ISFSI or the HSMs. There is no interaction between the stored empty canisters and the HSMs at the ISFSI. The ability of the modules to perform their physical protection, heat removal and shielding function is not affected by the presence of the stored empty canisters at the ISFSI site. In addition, storage of the DSCs will be in accordance with the plant criteria for storage of safety related components. Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the SAR is not increased.

___Yes _X_ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

No malfunctions are associated with temporary storage of empty canisters at the ISFSI site as described in the proposed activity.

____Yes _X_ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

The only potential accident associated with storage of empty canisters at the ISFSI site is the possible dislodging of the canisters such that one or more could roll towards an HSM that contains stored fuel and

ACTIVITY: Storage of Empty DSCs at Calvert Cliffs ISFSI **50.59 Log No.____ or 72.48 Log. No.** 94-0-101-002

block the inlet vents or damage the module by its impact. Since the empty canisters are oriented such that they would have to turn 90° to roll toward the modules, such an event is unlikely. Also, the possible contact angles between the canister and the module range from 0° to 90°. At 0° the canister contacts the module tangentially. At 90° the end of the canister contacts the module. Since the diameter of the canister is less than the width of the module inlet vent, there is no contact angle which allows the canister to completely block the module inlet vent. The probability of an accident evaluated in the ISFSI SAR is therefore not increased.

____Yes _X_ No May the consequences of an accident previously evaluated in the SAR be increased?

The consequences of the above stated potential accidents associated with storage of empty canisters at the ISFSI site are not increased for the following reasons:
a. If an empty canister finds its way to an HSM and partially blocks a vent, this condition is covered in the design basis analysis of the HSMs (Ref. USAR Section 8.2.7). The design basis analysis assumes that the vent is completely blocked up to 48 hours. Having a canister as the object blocking the vent does not affect the ability to move it within 48 hours. Such a condition will be identified within 24 hours by the required daily survailance of the ISFSI site.
b. The design basis for evaluating the HSM resistance to a massive impact load is a 3967 pound automobile with a 20 square foot frontal area traveling at a speed of 184.8 ft/sec impacting the side wall of an HSM. This results in a kinetic energy of 2,100,000 ft-lbs. To obtain the equivalent kinetic energy with a 34,330 pound empty canister would require a velocity of approximately 35 mph (Ref. BG&E calculation No. C-93-356). Such velocity is not possible to obtain since the DSCs are stored at approximately 30-150 feet away from the HSMs. It is, therefore, impossible for a DSC to turn 90° and accelerate to 35 mph across level gravel to impact the HSMs.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not increased.

____Yes _X_ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

There is no interaction between the empty canisters stored at the ISFSI site and the HSMs. Since the heavy weight of the empty canisters and the position of their storage does not allow them to accidentally roll and impact the HSMs, there is no possibility for a malfunction of a different type than any evaluated previously in the SAR being created.

____Yes _X_ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

The accidents considered in the SAR bound all potential accidental interactions between the stored empty canisters and the HSMs. No possibility of a new accident type is therefore created.

ATTACHMENT 3, SAFETY EVALUATION FORM

Page 4 of 4

ACTIVITY: **Storage of Empty DSCs at Calvert Cliffs ISFSI** **50.59 Log No._____ or 72.48 Log. No. 94-0-101-002**

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

____Yes _X_ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
N/A	No Technical Specifications are affected by the proposed activity

Complete for 72.48:

____Yes _X_ No Will the proposed activity involve a significant increase in occupational dose?

Table 7.4-1 of the ISFSI USAR Vol. I provides personnel dose estimates for fuel storage tasks. The task of storing and retrieving the empty DSCs from the ISFSI site will have negligible occupational dose since the DSCs are stored at a distance away from the location of the HSMs. Any occupational dose resulting from this activity is covered by the ISFSI USAR which allows daily inspection of the site by security personnel.

____Yes _X_ No Will the proposed activity involve a significant unreviewed environmental impact?

Because the conditions created by the storage of the empty canisters inside the ISFSI fenced area are bounded by the current safety analysis, this activity will not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

The site of the Independent Spent Fuel Storage Installation (ISFSI) is being used to store empty Dry Shielded Canisters (DSCs) horizontally on cribbing. The empty DSCs are positioned such that their ends are in the north-south direction facing the Horizontal Storage Modules (HSMs) where spent fuel is stored. The existing safety analysis documented in the ISFSI SAR bounds all possible interactions between the stored empty canisters and the HSMs at the ISFSI. These include the potential for the empty canisters to dislodge from their cribbing, roll towards the concrete modules and impact them or partially block the cooling vents that provide passive ventilation for decay heat removal from these modules. Therefore, the storage of empty DSCs inside the fenced security area of the ISFSI does not constitute an unreviewed safety question, a significant increase in occupational exposure, nor an unreviewed environmental impact for the ISFSI.

CCI-177
Rev. B/Change 0

ATTACHMENT 2, UFSAR CHANGE REQUEST FORM (UCR)

NONMOD # 94-28

To: UFSAR Coordinator
From: Sam Shakir (Printed Name) Work Group CCSO Date 8/4/94
Phone Number: 2179 System Number 101

SECTION 1 *(Change Initiation)*

UFSAR CHANGE SOURCE DOCUMENT

FCR/FEC/MCR # ______ (Circle One) Safety Evaluation Log # 94-0-101-02

RDC ______ Procedure # ______

License Amendment # ______

Regulatory Generic Correspondence # ______ (Generic Letter, Bulletin or Information Notice)

Unit 1 ____ Unit 2 ____ Common ____ ISFSI X

DESCRIPTION OF UFSAR CHANGE:

1) Add the following to the end of the second paragraph in Volume I, Section 4.1.1:
"The ISFSI site may be utilized for storage of empty DSCs. The empty DSCs may be stored there until they are needed for spent fuel loading and permanent storage. The empty DSCs will be stored horizontally on wood cribbing with their ends facing north-south at a distance from the HSMs to allow for normal spent fuel transportation and storage activities."

UFSAR SECTIONS AFFECTED: [Attach Marked up Page(s)]
Volume I, Section 4.1.1

CCI-177
Rev. B/Change 0

ATTACHMENT 2, UFSAR CHANGE REQUEST FORM (UCR)

SECTION 2 *(Interdisciplinary Reviews)*

RESP. IND. ______________ WORK GROUP: ______________
Printed Name and Signature
RESP. IND. ______________ WORK GROUP: ______________
Printed Name and Signature
RESP. IND. ______________ WORK GROUP: ______________
Printed Name and Signature

SECTION 3 *(Implementation Verification Prior to UFSAR Incorporation)*

VERIFICATION THAT PLANT MODIFICATION OR AS-BUILT INFORMATION HAS BEEN IMPLEMENTED:

☐ Partial Implementation

(For changes which have been partially implemented, identify the completed portion of the change on the marked-up UFSAR pages. If implementation is complete on one unit only, check the appropriate box, below.)

☐ Unit 1 ☐ Unit 2

RESPONSIBLE ENGINEER: ______________ DATE: ______

SECTION 4 *(Final Review/Approval Prior to UFSAR Incorporation)*

FINAL REVIEW & APPROVAL OF THIS CHANGE:

RESPONSIBLE ENGINEER: Sam Shakir SAM SHAKIR DATE: 8/4/94
RESP. ENGR'S. SUPERVISOR: Sam Shakir for M. Taylor PER TELE CON. DATE: 8/4/94
UFSAR COORDINATOR: ______________ DATE: ______
PE-LICENSING UNIT OR WGL: ______________ DATE: ______

4.0 INSTALLATION DESIGN

4.1 SUMMARY DESCRIPTION

4.1.1 LOCATION AND LAYOUT OF THE INSTALLATION

The location and layout of the Calvert Cliffs ISFSI with respect to other plant site structures is shown in Figure 4.1-1. This figure also denotes the route for transport of the transfer cask carrying DSCs from the Auxiliary Building to the ISFSI.

The initial construction phase of the ISFSI will include four 2x6 HSM arrays which will store up to 48 DSCs; each DSC contains 24 fuel assemblies. Additional HSM storage capacity will be added incrementally up to a total of ten 2x6 HSM arrays as needed. Figure 4.1-2 shows the arrangement of the storage arrays.

ADD ← SEE PG.1

The area around the ISFSI will be sloped to direct surface drainage to collection ditches for channeling rain water away from the site. As noted in Section 2.4, the ISFSI is about 86 feet above the probable maximum flood elevation. Local intense rainfall is not a problem since the resulting flood water would need to rise at least 18 inches above yard grade in order to block the HSM air inlets. (This height represents the bottom of the air inlet penetration on the inside of the air inlet plenum.) Adequate surface drainage exists at the ISFSI yard to assure that water will not collect to a depth of any concern.

The chosen transport route has been reviewed and is found to be in compliance with the design criteria of the transfer cask drop analysis discussed in Section 8.2 of the NUHOMS-24P Topical Report (Reference 4.1). Furthermore, the transport route has been reviewed to assure that no roadways, subgrade structures, buried pipes or trenches will be damaged by the transport trailer wheel loads. The approach slab has adequate space for turning the transport trailer and tow vehicle. No other turning areas are needed along the transport route.

4.1.2 PRINCIPAL FEATURES

4.1.2.1 Site Boundary

The property owned by BG&E surrounding the Calvert Cliffs ISFSI is shown in Figure 4.1-3.

4.1.2.2 Controlled Area [See Reference 4.5]

The controlled area for the ISFSI, as defined by 10 CFR 72.106, is identified in Figure 4.1-3. Its border from the HSM array is a minimum of 3900 feet (1189 meters) as shown in Figure 4.1-3.

4.1.2.3 Site Utility Supplies and Systems

No utility systems are required for the storage phase of the ISFSI. Electrical power will be provided to operate the hydraulic pumps used during DSC insertion or withdrawal operations at the HSM, and for lighting and security systems. No water or sewer systems are necessary. The existing plant page system will be extended to provide telephone and paging communications.

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 1

This screening is for: ______10 CFR 50.59 Applicability __x__ 10 CFR 72.48 Applicability
(Check one regulation only)

______CCNPP __x__ISFSI
(Check one facility only)

(Check one activity type only)

______**Procedure:** Procedure No./Change No.:__________________

______**Temporary Alteration:** Temporary Alteration No.:__________________

______**Setpoint Change:** SCAF No(s):__________________

______**Modification:** MCR/FCR/FEC No.:__________________
FEC Supplement No.:__________________

______

______**Core Reload:** Unit and Cycle:__________________

__X__**UFSAR/USAR:** UFSAR/USAR Change No.: 94-28

______**Other:** Identify Activity Type:__________________

Brief description of the activity:
The activity allows the storage of empty Dry Shielded Canisters (DSCs) horizontally on wood cribbing inside the security fence of the ISFSI site. The stored empty DSCs will be positioned such that their ends are in the north-south direction facing the Horizontal Storage Modules (HSMs) at a distance away from the HSMs to allow for normal spent fuel transportation and storage activities. The ISFSI site provides a secure and convenient storage area for the empty DSCs until they are loaded with spent fuel from the spent fuel pool and stored in the HSMs.

Technical Specifications/License Conditions (10 CFR 50.59/72.48)

1.____YES __x__NO Is the proposed activity a change or will it cause a change to the Technical Specifications/License Conditions or Bases?

2.____YES __x__NO Will the proposed activity cause Structures, Systems or Components (SSCs) to be operated in a manner that violates the Technical Specifications/License Conditions or Bases?

If both answers are "No," continue with the screening. Justification for each "No" answer shall be provided. List the sections of the Technical Specifications/License Conditions that were reviewed.
Justification:

There are no Tech. Spec. requirements that are violated by this activity, nor would the activity require a change to the ISFSI Tech. Spec. Storage of empty DSCs inside the ISFSI site will not affect the fuel handling and storage operation.

Technical Specifications/License Condition Sections Reviewed:

Reviewed all section of the ISFSI Tech. Spec.

If either of the above answers is "Yes," complete a Safety Evaluation and consult CCI-143 for License Amendment Proposals.

CCNPP/ISFSI Facility (10 CFR 50.59/72.48)

1.____YES __x__NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of the structure, system or component (SSC) directly affected by the activity?

If "No," answer each question below:

Why is the SAR description of the function of the SSC not affected?

The function of the DSCs is to provide mechanical confinement and containment for the stored spent fuel assemblies. DSCs loaded with spent fuel are inserted in the HSMs at the ISFSI site. Storage of the empty DSCs inside the fence at the ISFSI site occurs when the DSCs are not performing their intended function and, therefore, has no impact on their function.

Why is the SAR description of the method of performing the function of the SSC not affected?

Storage of the empty DSCs inside the fence at the ISFSI site occurs when the DSCs are not performing their intended function. The DSCs perform their function by providing confinement for the spent fuel assemblies in a sealed environment, so the spent fuel can be transferred from the Auxiliary Building to the ISFSI and stored inside the Horizontal Storage Modules. Therefore, storing the empty DSCs before they are utilized for fuel storage has no affect on the way these DSCs perform their function.

Why is the SAR description of the design of the SSC not changed?

The DSCs are high integrity stainless steel, welded pressure vessels that provide confinement for the stored fuel assemblies. The DSCs are designed to provide radiological shielding and physical protection during the loading operation and storage. Allowing some empty DSCs to be stored inside the ISFSI site, when they are not performing their intended function, has no impact on the design of these components.

2.____YES __x__NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of any other SSC described in the SAR?

If "No," answer the following question:

Explain why the activity does not affect other SSCs described in the SAR..

Storage of the empty DSCs does not affect the HSMs located in the ISFSI site. The empty DSCs will be stored such that the long axis of their cylindrical body is perpendicular to the face of the HSMs, and at a distance away from the HSMs enough to allow normal spent fuel transportation and loading activities. There is no interaction between the empty canisters and the HSMs. The heavy weight of the canisters and the position of their storage does not allow them to accidentally roll and impact the HSMs. No other SSCs are affected by this activity.

3.__x__YES ______NO Is the proposed activity a revision to the SAR. (Editorial changes are limited to obvious grammatical/spelling errors, reorganization of portions of the SAR or minor changes that do not affect the intent of the information conveyed by a drawing.)

4.__x__YES ______NO Will the proposed activity add to or delete from the SAR description of a SSC?

Procedures (10 CFR 50.59/72.48)

1.____YES __x__NO Will the proposed activity affect the intent of any procedure described in the SAR (editorial changes do not need a Safety Evaluation)? The NRC staff does not consider procedures simply listed in the SAR to be described in the SAR. Also, procedures include anything that defines or describes activities or controls over functions, tasks, reviews, tests and safety review meetings.

2._____YES __x__NO Will the proposed activity cause SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: The storage of empty DSCs in the ISFSI site does not affect any procedures described in the ISFSI USAR. Storing the empty DSCs when they are not performing their intended function has no impact on their design, function, or method of performing their function as described in the ISFSI USAR.

Tests or Experiments (10 CFR 50.59/72.48)

1._____YES __x___NO Will the proposed activity result in conducting a test or experiment causing SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: This activity is not a test or experiment.

ISFSI (10 CFR 72.48) These questions are only required to be answered for activities affecting ISFSI.

1._____YES ___x___NO Will the proposed activity increase any occupational dose for ISFSI related activities?

2._____YES ___x___NO Will the proposed activity use additional property for ISFSI operations?

3._____YES ___x___NO Will the proposed activity add or change the roads or transport equipment, including cranes, used for ISFSI operations?

Justify each "No" answer below:

Justification: Storage and retrieval of the empty DSCs from the ISFSI site does not affect the occupational dose for ISFSI related activities, nor does it impact the spent fuel storage operation. Storage of the DSC's will be inside the ISFSI security fence and will not use any additional property, change the roads, or change the transport equipment.

SAR Sections Reviewed:

Volumes I & IV of the ISFSI USAR

If **ALL** answers are "No", A Safety Evaluation is not required.

If **ANY** answer is "Yes", A Safety Evaluation is required.

1._____YES __x__NO Does this activity require additional screening?

10CFR 50.59 For Impact on CCNPP
10 CFR 72.48 For Impact on ISFSI

If "Yes", Perform a separate Safety Evaluation Screening.

Prepared By: SAM SHAKIR Sam Shakir Date: 8/4/94
PRINTED NAME AND SIGNATURE

EN-1-102
Revision 1

ATTACHMENT 3, SAFETY EVALUATION FORM

Page 1 of 4

ACTIVITY: Calvert Cliffs ISFSI USAR Change 50.59 Log No.____ or 72.48 Log No. 94-0-101-003

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

___YES X NO Involve an Unreviewed Safety Question (USQ)?
___YES X NO Involve a change to the Technical Specifications/License Conditions or Bases?
X YES ___ NO Require a change or addition to the UFSAR or USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

___YES X NO Involve a Significant Increase in Occupational Dose?
___YES X NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Sam Shakir SAM SHAKIR Department: CCSO Date: 8/2/94
PRINTED NAME AND SIGNATURE (VECTRA)

X YES __ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: J. B. MAKAR PRINTED NAME	Resp. Ind.: G. Tesfaye PRINTED NAME	Resp. Ind.:____ PRINTED NAME
J B Mak SIGNATURE	Getachew Tesfaye SIGNATURE	____ SIGNATURE
Work Group: System Engineer	Work Group: Licensing Unit	Work Group:____
Date: 8-5-94	Date: 9-20-94	Date:____

Approved ✓ Disapproved ___	Approved ✓ Disapproved ___
Signature Sam Shakir for Moses Taylor INDEPENDENT REVIEWER (VECTRA) Per Tele. Con.	Signature Michael J. Gahan [✓ GS-DES] GS-TSES, or PE-PDSU M. J. GAHAN III CCGU 8/10/94 9.20.94
Date: 8/4/94	Date:____

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 94-145 Date: 9-28-94

Recommend Approval ✓ Recommend Disapproval____ Signature____ Date: 9-28-94
POSRC CHAIRMAN

Approved ✓ Disapproved____ Signature____ Date: 9/28/94
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
OSSRC Meeting No.: 95-003 Date:____

Recommend Approval____ Recommend Disapproval____ Signature____ Date:____
OSSRC CHAIRMAN

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: Calvert Cliffs ISFSI USAR Change 50.59 Log No.____ or 72.48 Log No. 94-0-101-003

Proposed Activity:
This activity changes the requirements for the ISFSI transfer route to allow the shoulders to be up to 20" lower than the centerline elevation of the road surface. This activity results in changing the ISFSI USAR as follows:

1) Change USAR Volume IV, Section 2 USAR Q&A, Question 8.0-5 Response, first paragraph to read:

"The transfer cask will be transported along an asphalt or concrete paved road which is at least 16 feet wide and which has shoulders which extend to make the transfer route at least 28 feet wide. The road is approximately 3,300 linear feet with grades which range from 0% to 3% except for an approximate 50 foot length which carries a 5.7% grade. The roadbed is level except for a negligible 1% slope required to create a crown in the road for drainage and a transverse slope at any point along the transportation route of less than 10%. The shoulders are either level with the road, or slope down from the road such that the maximum vertical distance from the centerline of the road to the lowest point within the 28 foot wide transfer route is 20 inches. In those locations where the paved road abuts up to existing blacktop, or concrete paving, the shoulder is discontinued. The shoulder may be paved, gravel or soil and contain typical roadside fixtures, including curbs, fences, guard rails and light poles which do not constitute potential puncture mechanisms for the cask during a drop. The shoulders do not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture mechanism during a cask drop. For the entire route that the transfer cask is transported there will exist a minimum 8 foot wide zone on each side of the trailer that is not more than 20 inches below the road centerline elevation."

2) Change USAR Volume I, Section 10.3.4.1, Item B. Specifications, first paragraph to read:

"The roadway or ground surface elevation perpendicular to the route to or from the ISFSI within an 8.0 ft proximity of the transfer trailer shall not be more than 20 inches below the trailer road surface centerline elevation. The paved portion of the road shall be a minimum of 16 feet wide and the adjacent paved, gravel or soil shoulder shall extend to make the transfer route at least 28 feet wide. The lowest point within the 28 foot wide transfer route shall not be lower than 20 inches below the road centerline and may contain typical roadside fixtures, including curbs, fences, guard rails and light poles which do not constitute potential puncture mechanisms for the cask. The shoulders may not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture mechanism. The road shall be closed to other vehicles when transporting the spent fuel."

Reason for Activity:
The current ISFSI USAR description of the transfer route and shoulders is unnecessarily restrictive regarding the allowable elevation of the shoulder surface relative to the transfer road surface and the relative width of the paved road and the adjacent shoulders. The current description of the road specifies the elevation of the shoulder surface to be not less than that of the trailer road surface centerline elevation. This description is restrictive considering that the shoulders are affected by heavy rain and at times get eroded and washed away requiring constant repair. The significance of the shoulder elevation is to limit the drop height of the cask to its designed limit of 80 inches. Since the maximum distance from the bottom of the transfer cask to the road centerline is 56.25 inches, this allows the lowest point on the transfer route to be up to 20 inches below the elevation of the road centerline without affecting the design basis of 80 inches. The current description of the shoulders width is also restrictive. The ISFSI USAR describes the shoulders as being a minimum of 7 feet wide on each side of the road. This will now be changed to specify a total width of the transfer route including shoulders at a minimum of 28 feet.

Function (s) of affected SSC:
Transport road provides a hard paved surface for the tractor to transport spent fuel in a NUHOMS®-24P canister/transfer cask from the Auxiliary Building to the ISFSI.

ISFSI USAR Sections Reviewed:
Vol.. IV, Section 2; Vol. I, Section 4.1.1; Vol. I, Section 10.3

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: Calvert Cliffs ISFSI USAR Change 50.59 Log No.____ or 72.48 Log No. 94-0-101-003

Complete for 50.59 and 72.48:

1. The probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

___Yes _X_ No May the probability of ocurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

The function of the canister and cask during transfer operations is not affected by the proposed changes since they do not cause the cask to exceed the design basis drop height of 80 inches. (Ref. BG&E Calc. C-91-75, C-91-76, & C-93-325)

___Yes _X_ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

The consequences of a malfunction are not affected by the proposed changes since there are no malfunctions associated with these changes.

___Yes _X_ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

The probability of a drop accident from above the 80 inches design basis drop height is not increased because the physical dimensions of the cask and trailer and associated transport equipment prevent the cask from exceeding a height of 80 inches if the maximum difference in elevation from the centerline of the road and lowest point on the shoulder is limited to 20 inches. Drop accidents for a Dry Shielded Canister (DSC) loaded with fuel in a transfer cask have been analyzed and can be sustained without unacceptable damage to the cask and DSC for heights up to 80 inches above a thick hard surface.

___Yes _X_ No May the consequences of an accident previously evaluated in the SAR be increased?

No accidents or consequences are associated with the proposed changes in allowable transportation route configuration since the proposed changes do not cause the cask to exceed the design basis drop accident height of 80 inches.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not increased.

___Yes _X_ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Any malfunction of the transfer cask would be associated with a drop from a height greater than 80 inches. Since the proposed changes do not result in this condition, the possibility of a new malfunction is not created.

___Yes _X_ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

The proposed changes affect transport of spent fuel inside the Dry Shielded Canister using the transfer cask, an analyzed condition. Since the bounding case envelopes the proposed activities, no possibility of a new accident is created.

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: Calvert Cliffs ISFSI USAR Change 50.59 Log No.____ or 72.48 Log No. 94-0-101-003

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

____Yes _X_ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
2.3	Section 2.3 states that the Transfer Cask lifting height outside the Auxiliary Building shall not exceed 80 inches. In addition, in the event of a transfer cask drop from a height greater than 15 inches, action to inspect must be taken.

The maximum distance from the bottom of the transfer cask to the road centerline is 56.25 inches. Allowing the lowest point on the transfer route to be up to 20 inches below the elevation of the road centerline would limit the possible drop height for the cask to 76.25 inches which is below the design basis 80 inches.

Complete for 72.48:

____Yes _X_ No Will the proposed activity involve a significant increase in occupational dose?

The proposed changes do not cause the transfer cask to be placed in an unanalyzed condition. They do not therefore affect the occupational exposure for the ISFSI.

____Yes _X_ No Will the proposed activity involve a significant unreviewed environmental impact?

Since the transfer route road and shoulder configuration as described by the proposed changes is bounded by the current safety analysis, it does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

A transport road provides a hard paved surface for a tractor to transport spent fuel in a NUHOMS®-24P canister/transfer cask from the Auxiliary Building to the Independent Spent Fuel Storage Installation (ISFSI).
The ISFSI USAR description of the transfer route road and shoulders was changed to avoid being unnecessarily restrictive regarding the allowable elevation of the shoulder surface relative to the transfer road surface and the relative width of the paved road and the adjacent shoulders. The proposed change allows the road shoulder surface within the 28 foot wide transfer route to be up to 20 inches below the road centerline rather than at or above the road surface. The proposed change also specifies the road configuration in terms of minimum requirements for the relative width of road and shoulder surfaces rather than specific relative widths. This change does not constitute an unreviewed safety question, a change to the Technical Specifications or Bases, a significant increase in occupational exposure or an unreviewed environmental impact for the ISFSI.

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 1

This screening is for: ______**10 CFR 50.59 Applicability** __x__ **10 CFR 72.48 Applicability**
(Check one regulation only)

______**CCNPP** __x__**ISFSI**
(Check one facility only)

(Check one activity type only)

______**Procedure:**	Procedure No./Change No.:______
______**Temporary Alteration:**	Temporary Alteration No.:______
______**Setpoint Change:**	SCAF No(s):______
______**Modification:**	MCR/FCR/FEC No.:______ FEC Supplement No.:______
______**Core Reload:**	Unit and Cycle:______
__x__**UFSAR/USAR:**	UFSAR/USAR Change No.: 94-30
______**Other:**	Identify Activity Type:______

Brief description of the activity:

Change the ISFSI USAR current description of the transfer route and shoulders which is unnecessarily restrictive regarding the allowable elevation of the shoulder surface relative to the transfer road surface and the relative width of the paved road and adjacent shoulders. The route is used for transporting the cask/canister assembly between the Auxiliary Building and the ISFSI.

Technical Specifications/License Conditions (10 CFR 50.59/72.48)

1._____YES __x__NO Is the proposed activity a change or will it cause a change to the Technical Specifications/License Conditions or Bases?

2._____YES __x__NO Will the proposed activity cause Structures, Systems or Components (SSCs) to be operated in a manner that violates the Technical Specifications/License Conditions or Bases?

If both answers are **"No,"** continue with the screening. Justification for each **"No"** answer shall be provided. List the sections of the Technical Specifications/License Conditions that were reviewed.
Justification:

Changing the ISFSI USAR description of the transfer road does not affect any technical specification.

Technical Specifications/License Condition Sections Reviewed:

Reviewed all sections of the ISFSI Technical Specification manual.

If either of the above answers is **"Yes,"** complete a Safety Evaluation and consult CCI-143 for License Amendment Proposals.

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 2

CCNPP/ISFSI Facility (10 CFR 50.59/72.48)

1.__x__YES _____NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of the structure, system or component (SSC) directly affected by the activity?

If "No," answer each question below:

Why is the SAR description of the function of the SSC not affected?

Why is the SAR description of the method of performing the function of the SSC not affected?

Why is the SAR description of the design of the SSC not changed?

2._____YES ___x___NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of any other SSC described in the SAR?

If "No," answer the following question:

Explain why the activity does not affect other SSCs described in the SAR..

Changing the ISFSI USAR description of the transfer road does not affect other SSCs in the plant or the ISFSI.

3.__x__YES ______NO Is the proposed activity a revision to the SAR. (Editorial changes are limited to obvious grammatical/spelling errors, reorganization of portions of the SAR or minor changes that do not affect the intent of the information conveyed by a drawing.)

4.__x__YES ______NO Will the proposed activity add to or delete from the SAR description of a SSC?

Procedures (10 CFR 50.59/72.48)

1._____YES ___x__NO Will the proposed activity affect the intent of any procedure described in the SAR (editorial changes do not need a Safety Evaluation)? The NRC staff does not consider procedures simply listed in the SAR to be described in the SAR. Also, procedures include anything that defines or describes activities or controls over functions, tasks, reviews, tests and safety review meetings.

2._____YES ___x__NO Will the proposed activity cause SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: The activity changes the description of the transfer route in the ISFSI USAR and does not affect any procedures or change the method of transporting the cask between the Auxiliary Building and the ISFSI.

Tests or Experiments (10 CFR 50.59/72.48)

1._____YES ___x___NO Will the proposed activity result in conducting a test or experiment causing SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: This activity is not a test or experiment.

ISFSI (10 CFR 72.48) These questions are only required to be answered for activities affecting ISFSI.

1._____YES ___x___NO Will the proposed activity increase any occupational dose for ISFSI related activities?

2._____YES ___x___NO Will the proposed activity use additional property for ISFSI operations?

3.__x__YES ______NO Will the proposed activity add or change the roads or transport equipment, including cranes, used for ISFSI operations?

Justify each "No" answer below:

Justification: Changing the road description in the ISFSI USAR does not impact the method of performing the transport and storage operation of the spent fuel and therefore, does not increase the occupational dose for any of the ISFSI related activities nor does it require the use of additional property for ISFSI operations.

SAR Sections Reviewed:

Volumes I & IV of the ISFSI USAR

If **ALL** answers are "No", A Safety Evaluation is not required.

If **ANY** answer is "Yes", A Safety Evaluation is required.

1.___x___YES ______NO Does this activity require additional screening?

10CFR 50.59 For Impact on CCNPP
10 CFR 72.48 For Impact on ISFSI

If "Yes", Perform a separate Safety Evaluation Screening.

Prepared By: SAM SHAKIR Sam Shakir Date: 8/4/94

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 1

This screening is for: __x__ **10 CFR 50.59 Applicability** ____ **10 CFR 72.48 Applicability**
(Check one regulation only)

__x__ **CCNPP** _____ **ISFSI**
(Check one facility only)

(Check one activity type only)

______ **Procedure:** Procedure No./Change No.:__________

______ **Temporary Alteration:** Temporary Alteration No.:__________

______ **Setpoint Change:** SCAF No(s):__________

______ **Modification:** MCR/FCR/FEC No.:__________
FEC Supplement No.:__________

______ **Core Reload:** Unit and Cycle:__________

__x__ **UFSAR/USAR:** UFSAR/USAR Change No.: 94-30

______ **Other:** Identify Activity Type:__________

Brief description of the activity:

Change the ISFSI USAR current description of the transfer route and shoulders which is unnecessarily restrictive regarding the allowable elevation of the shoulder surface relative to the transfer road surface and the relative width of the paved road and adjacent shoulders. The route is used for transporting the cask/canister assembly between the Auxiliary Building and the ISFSI.

Technical Specifications/License Conditions (10 CFR 50.59/72.48)

1._____YES __x__NO Is the proposed activity a change or will it cause a change to the Technical Specifications/License Conditions or Bases?

2._____YES __x__NO Will the proposed activity cause Structures, Systems or Components (SSCs) to be operated in a manner that violates the Technical Specifications/License Conditions or Bases?

If both answers are **"No,"** continue with the screening. Justification for each **"No"** answer shall be provided. List the sections of the Technical Specifications/License Conditions that were reviewed.
Justification:

Changing the ISFSI USAR description of the transfer road does not affect any technical specification. No sections in the U1 or U2 Tech. Spec. is applicable.

Technical Specifications/License Condition Sections Reviewed:

Reviewed all sections of the U1 and U2 Tech. Spec.

If either of the above answers is **"Yes,"** complete a Safety Evaluation and consult CCI-143 for License Amendment Proposals.

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 2

CCNPP/ISFSI Facility (10 CFR 50.59/72.48)

1._____YES __x__NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of the structure, system or component (SSC) directly affected by the activity?

If "No," answer each question below:

Why is the SAR description of the function of the SSC not affected?

The affected SSC is the transport road which provides a hard paved surface to transport the NUHOMS®-24P canister/transfer cask from the Auxiliary Building to the ISFSI. This activity does not affect this described function. The description of this road is only included in the ISFSI USAR and not in the UFSAR.

Why is the SAR description of the method of performing the function of the SSC not affected?

The UFSAR has no description of the transport road from the Auxiliary to the ISFSI. This activity changes the road's description in the ISFSI USAR (see 72.48 evaluation log No. 94-0-101-003) and does not affect the function or the method of performing the function of the road.

Why is the SAR description of the design of the SSC not changed?

The road is designed to withstand the loads from the tractor that transports the canister/transfer cask assembly from the Auxiliary Building to the ISFSI. The description of the road design exists in the ISFSI USAR only and not in the UFSAR (see 72.48 evaluation log No. 94-0-101-003). No other design description is affected by this activity.

2.____YES ___x___NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of any other SSC described in the SAR?

If "No," answer the following question:

Explain why the activity does not affect other SSCs described in the SAR..

This activity changes the ISFSI road description provided only in the ISFSI USAR. It does not affect the function or the method of performing the function of the road or any other SSCs described in the SAR.

3.____YES ___x___NO Is the proposed activity a revision to the SAR. (Editorial changes are limited to obvious grammatical/spelling errors, reorganization of portions of the SAR or minor changes that do not affect the intent of the information conveyed by a drawing.)

4._____YES ___x___NO Will the proposed activity add to or delete from the SAR description of a SSC?

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 3

Procedures (10 CFR 50.59/72.48)

1.____YES ___x___NO Will the proposed activity affect the intent of any procedure described in the SAR (editorial changes do not need a Safety Evaluation)? The NRC staff does not consider procedures simply listed in the SAR to be described in the SAR. Also, procedures include anything that defines or describes activities or controls over functions, tasks, reviews, tests and safety review meetings.

2.____YES ___x___NO Will the proposed activity cause SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: This activity does not affect any SSCs described in the UFSAR. The transfer of fuel from the Auxiliary Building to the ISFSI is outlined in the ISFSI USAR (see 72.48 evaluation No. 94-0-101-003). Changing the description of the road in the ISFSI USAR does not affect any procedures or the method of transporting the fuel on the road.

Tests or Experiments (10 CFR 50.59/72.48)

1.____YES ___x___NO Will the proposed activity result in conducting a test or experiment causing SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: This activity is not a test or experiment.

ISFSI (10 CFR 72.48) These questions are only required to be answered for activities affecting ISFSI.

1.____YES _____NO Will the proposed activity increase any occupational dose for ISFSI related activities?

2.____YES _____NO Will the proposed activity use additional property for ISFSI operations?

3.____YES ______NO Will the proposed activity add or change the roads or transport equipment, including cranes, used for ISFSI operations?

Justify each "No" answer below:

Justification:

SAR Sections Reviewed:

Volumes I & IV of the ISFSI USAR

If <u>**ALL**</u> answers are "No", A Safety Evaluation is not required.

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 4

If <u>ANY</u> answer is "Yes", A Safety Evaluation is required.

1. __x__ YES ______ NO Does this activity require additional screening?

10CFR 50.59 For Impact on CCNPP
10 CFR 72.48 For Impact on ISFSI

If "Yes", Perform a separate Safety Evaluation Screening.

Prepared By: SAM SHAKIR Sam Shakir Date: 8/4/94
PRINTED NAME AND SIGNATURE

ATTACHMENT 2, UFSAR CHANGE REQUEST FORM (UCR)

NONMOD # 94-30

To: UFSAR Coordinator
From: SAM SHAKIR (PRINTED NAME) Work Group CCSO Date 8/03/94
Phone Number: 2179 System Number 101

SECTION 1 (Change Initiation)

UFSAR CHANGE SOURCE DOCUMENT

FCR/FEC/MCR # ______ (Circle One)
Safety Evaluation Log # 94-0-101-003
RDC ______
Procedure # ______
License Amendment # ______
Regulatory Generic Correspondence # ______ (Generic Letter, Bulletin or Information Notice)

Unit 1 ____ Unit 2 ____ Common ____ ISFSI X

DESCRIPTION OF UFSAR CHANGE:

1) Change Volume IV, Section 2 ISFSI USAR Q&A, Question 8.0-5 Response, first paragraph to read:

"The transfer cask will be transported along an asphalt or concrete paved road which is at least 16 feet wide and which has shoulders which extend to make the transfer route at least 28 feet wide. The road is approximately 3,300 linear feet with grades which range from 0% to 3% except for an approximate 50 foot length which carries a 5.7% grade. The roadbed is level except for a negligible 1% slope required to create a crown in the road for drainage and a transverse slope at any point along the transportation route of less than 10%. The shoulders are either level with the road, or slope down from the road such that the maximum vertical distance from the centerline of the road to the lowest point within the 28 foot wide transfer route is 20 inches. In those locations where the paved road abuts up to existing blacktop, or concrete paving, the shoulder is discontinued. The shoulder may be paved, gravel or soil and contain typical roadside fixtures, including curbs, fences, guard rails and light poles which do not constitute potential puncture mechanisms for the cask during a drop. The shoulders do not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture mechanism during a cask drop. For the entire route that the transfer cask is transported there will exist a minimum 8 foot wide zone on each side of the trailer that is not more than 20 inches below the road centerline elevation."

ATTACHMENT 2, UFSAR CHANGE REQUEST FORM (UCR)

94-30

2) Change Volume I, ISFSI USAR Section 10.3.4.1, Item B. Specifications, first paragraph to read:

"The roadway or ground surface elevation perpendicular to the route to or from the ISFSI within an 8.0 ft proximity of the transfer trailer shall not be more than 20 inches below the trailer road surface centerline elevation. The paved portion of the road shall be a minimum of 16 feet wide and the adjacent paved, gravel or soil shoulder shall extend to make the transfer route at least 28 feet wide. The lowest point within the 28 foot wide transfer route shall not be lower than 20 inches below the road centerline and may contain typical roadside fixtures, including curbs, fences, guard rails and light poles which do not constitute potential puncture mechanisms for the cask. The shoulders may not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture mechanism. The road shall be closed to other vehicles when transporting the spent fuel."

UFSAR SECTIONS AFFECTED: (Attach Marked up Page(s))
ISFSI USAR Volume IV, Section 2 SAR Q&A, Question 8.0-5 Response, first paragraph
ISFSI USAR Volume I, Section 10.3.4.1, Item B. Specifications, first paragraph

CCI-177
Rev. B/Change 0

ATTACHMENT 2, UFSAR CHANGE REQUEST FORM (UCR)

94-30

SECTION 2 (Interdisciplinary Reviews)

RESP. IND. ______________________ WORK GROUP: ______________________
Printed Name and Signature
RESP. IND. ______________________ WORK GROUP: ______________________
Printed Name and Signature
RESP. IND. ______________________ WORK GROUP: ______________________
Printed Name and Signature

SECTION 3 (Implementation Verification Prior to UFSAR Incorporation)

VERIFICATION THAT PLANT MODIFICATION OR AS-BUILT INFORMATION HAS BEEN IMPLEMENTED:

☐ Partial Implementation

(For changes which have been partially implemented, identify the completed portion of the change on the marked-up UFSAR pages. If implementation is complete on one unit only, check the appropriate box, below.)

☐ Unit 1 ☐ Unit 2

RESPONSIBLE ENGINEER: ______________________ DATE: __________

SECTION 4 (Final Review/Approval Prior to UFSAR Incorporation)

FINAL REVIEW & APPROVAL OF THIS CHANGE:

RESPONSIBLE ENGINEER: Sam Shakir DATE: 8/03/94
RESP. ENGR'S. SUPERVISOR: Sam Shakir for M. Taylor Per Telecon DATE: 8/04/94
UFSAR COORDINATOR: ______________________ DATE: __________
PE-LICENSING UNIT OR WGL: ______________________ DATE: __________

RESPONSE TO NRC COMMENTS ON THE CALVERT CLIFFS NUCLEAR POWER PLANT ISFSI SAR

Section 8

QUESTION: 8.0-5

Para 8.2.5.

As stated in Section 2.1.1.1 of the CCNPP ISFSI ER, the minimum elevation difference between the ISFSI site and the plant site is 70 feet. Although statements are made in Sections 4.1.1 and 10.3.4.1 regarding the acceptability of the transportation route for the TC, provide more details on this road with specifics on the grading around the road and special provisions to ensure that the TC is not dropped greater than the 80 inches analyzed in the SAR during its transport over a 70 feet elevation gradient to the ISFSI site. What provisions will be made during the transport of the DSC to preclude the TC from rolling backwards on the slopped portion of the route in the event that the engine and brakes of the prime moving vehicle fail?

RESPONSE: (Revised by a 10 CFR 72.48 Safety Evaluation Process; Pacific Nuclear File Nos. BG001.0051.01 and BG001.0051.03.)

CHANGE

The transfer cask will be transported along an asphalt or concrete paved road which is 16 feet wide and has 7 to 8 feet shoulders. The road is approximately 3,300 linear feet with slopes which range from 0% to 3% except for an approximate 50 feet length which carries a 5.7% slope. The roadbed is level except for a negligible 1% slope required to create a crown in the road for drainage and a transverse slope at any point along the transportation route of less than 10%. The shoulders are either level with the road or slope up from the road. In those locations where the paved road abuts up to existing blacktop, or concrete paving, the shoulder is discontinued. The shoulder may be paved, gravel or soil and contain typical roadside fixtures, including curbs, fences, guard rails and light poles which do not constitute potential puncture devices for the cask during a drop. The shoulders do not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture device during a cask drop. For the entire route that the transfer cask is transported there will exist a minimum 8 feet wide zone that is at or above the roadbed elevation.

The transfer trailer braking system is not operable independent of the prime mover. However, failure of the prime mover will cause the trailer braking system to fail-safe, that is "lock tight".

10.3.4 LIMITING AND OPERATING CONDITIONS FOR TRANSFER CASK CONTAINING LOADED DSC

10.3.4.1 Transfer Route Selection [See Reference 10.2]

A. Title: Transfer Route Selection

CHANGE

B. Specifications: The roadway or ground surface elevation perpendicular to the route to or from the ISFSI within an 8.0 ft proximity of the transfer trailer shall not be less than that of the trailer road surface elevation as measured at the outer edge of asphalt pavement. The paved portion of the road shall be a minimum of 16 feet wide and the adjacent paved, gravel or soil shoulder shall be a minimum of 7 feet wide on each side of the road. The shoulder shall be level with or higher than the outer edge of the pavement and may contain typical roadside fixtures, including curbs, fences, guard rails and light poles which do not constitute potential puncture devices for the cask. The shoulders may not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture device. The road shall be closed to other vehicles when transporting the spent fuel.

The maximum drop height of the cask from the transfer trailer to the roadbed does not exceed 80 inches.

C. Applicability: This specification is applicable to DSC transfer utilizing the NUHOMS-24P transfer cask and trailer.

D. Objective: Ensure that a potential drop height of 80 inches is not exceeded.

E. Action: Repair the road to its proper elevation.

F. Surveillance: Prior to the transfer of a DSC to or from an HSM, the proposed transfer route shall be visually inspected.

G. Bases: A drop from a height of 80 inches or less does not compromise the design margins of the transfer cask or DSC.

ATTACHMENT 3, SAFETY EVALUATION FORM

Page 1 of 5

ACTIVITY: Calvert Cliffs ISFSI USAR Change 50.59 Log No.____ or 72.48 Log No. 94-0-101-004

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

___YES _X_ NO Involve an Unreviewed Safety Question (USQ)?
___YES _X_ NO Involve a change to the Technical Specifications/License Conditions or Bases?
_X_YES ___ NO Require a change or addition to the UFSAR or USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

___YES _X_ NO Involve a Significant Increase in Occupational Dose?
___YES _X_ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Sam Shakir Sam Shakir Department: CCSO Date: 7/13/94
PRINTED NAME AND SIGNATURE (VECTRA)

_X_YES ___ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: Robert H. Beall PRINTED NAME	Resp. Ind.: PRINTED NAME	Resp. Ind.: PRINTED NAME
Robert H. Beall SIGNATURE	SIGNATURE	SIGNATURE
Work Group: Fuel Management	Work Group:	Work Group:
Date: 7/13/94	Date:	Date:

Approved ✓ Disapproved ___	Approved ✓ Disapproved ___
Signature Sam Shakir FOR IAN McGINNIS PER TELECON. INDEPENDENT REVIEWER (VECTRA)	Signature Michael J. Gahan 7.13.94 GS-DES, GS-TSES, or PE-PDSU FOR [illegible] 7/13/94 CCSO
Date 7/13/94	Date

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 94-118 Date: 7-13-94

Recommend Approval ✓ Recommend Disapproval___ Signature___ Date: 7-13-94
POSRC CHAIRMAN

Approved ✓ Disapproved___ Signature___ Date: 7/13/94
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
OSSRC Meeting No.: 95-003 Date:___

Recommend Approval___ Recommend Disapproval___ Signature___ Date:___
OSSRC CHAIRMAN

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: Calvert Cliffs ISFSI USAR Change 50.59 Log No.____ or 72.48 Log No. 94-0-101-004

Proposed Activity:
This activity will support the new ISFSI fuel loading procedure (ISFSI-01) to allow the use of pressurized air or helium for liquid removal from the DSC cavity during the DSC drying operation. The vendor Tech. Manual already allows the use of either air or helium for this operation. This change will require the following ISFSI UFSAR changes:

1) Change Volume I, Section 1.3.1.7 to read:
"The vacuum drying system removes water and air from the DSC and fills it with helium. The vacuum drying system has four operational modes: water removal, helium or air forced water removal, vacuum pumping, and helium backfilling."

2) Change Volume I, Section 1.3.1.9 item I. to read:
"Air or helium lines are connected to the DSC vent port and the water inside the canister is forced out the siphon tube by pressurized air or helium."

3) Change Volume I, Section 4.3.1 to read:
"The VDS is designed to operate in four modes: liquid removal by pump, liquid removal by a source of pressurized helium or air, vacuum drying, and helium backfill. The evacuation is performed.........still present in the DSC."

4) Change Volume I, Section 5.1.1.3 to read:
"Connect the VDS to the DSC. Open the cask drain port valve and remove the remaining water from the cask/DSC annulus. Remove the remaining water from the DSC cavity by engaging the compressed helium supply or a compressed air source through the helium inlet connection and opening the valve to the DSC vent port, forcing the water from the DSC through the siphon port."

Reason for Activity:
To allow the use of pressurized air or helium for liquid removal from the ISFSI Dry Shielded Canister (DSC) by the Vacuum Drying System (VDS). The drying operation of the DSC using the VDS is carried out in four stages. The first stage removes liquid from the DSC by pumping. The second stage removes the remaining liquid from the DSC by pressurization using a compressed gas. The third stage is to vacuum dry the DSC, and the fourth and final stage is to backfill the DSC with helium. The change only affects the second stage of the operation where a large quantity of compressed gas is needed to remove the remaining liquid from the DSC. Permitting the use of pressurized air has two benefits. First, it will save a significant amount of helium needed for the blowdown of liquid, and second it will not release this volume of helium into the atmosphere of the surrounding Spent Fuel Pool area. The increased helium concentration may be detected by the helium leak detector used for measuring leakage from the DSC inner cover plate closure weld. The presence of helium in the air could result in a delay of the final acceptance of the DSC closure operation until the helium concentration is removed by the Auxiliary Building ventilation system.

Function (s) of affected SSC:
The DSC provides containment and confinement of the spent fuel during storage. The drying operation of the DSC using the VDS, provides the appropriate atmospheric environment for long term dry fuel storage in the DSC. The DSC is classified as Safety Related. The VDS provides a means for removing water and air from the DSC and for backfilling the DSC with helium. This function is required to ensure that fuel is stored in an inert atmosphere, and to take advantage of the heat transfer properties of helium. The VDS is classified as NSR.

ISFSI USAR Sections Reviewed:
Vol. I, Section 1.3.1.7, Vol. I, Section 1.3.1.9, Vol. I, Section 3.1.2.3, Vol. I, Section 4.3.1, Vol. I, Section 5.1.1.3. , ISFSI Tech. Spec. Section 2.2

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: Calvert Cliffs ISFSI USAR Change **50.59 Log No._____ or 72.48 Log No.** 94-0-101-004
Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

___Yes _X_ No May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

The function of the Dry Shielded Canister as a containment and confinement barrier is not affected by the use of pressurized air in lieu of compressed helium during liquid removal from the DSC. The pressurized air will perform the same function as compressed helium to force the liquid out of the DSC, and to prepare the DSC for the following two final stages of vacuum drying and helium backfilling. Therefore, the probability of a malfunction is not increased by the proposed change.

___Yes _X_ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

The consequences of a malfunction are not affected by the proposed changes since there are no malfunctions associated with these changes. The presence of air inside the DSC cavity for the short duration of the DSC drying operation will not cause any corrosive activity or degradation in the fuel cladding. The air will be removed from the DSC and replaced with helium by the VDS prior to full closure of the DSC to provide the required inert environment for long term dry storage of the fuel. There are no safety concerns associated with the malfunction of the non safety related VDS. A malfunction of the VDS will only result in a delay of the the DSC closure operation.

___Yes _X_ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

The probability of an accident in which the containment and confinement boundary formed by the DSC is breached is not increased by the proposed change. The use of pressurized air or helium to force the liquid out of the DSC during the drying operation is not relevant to the probability of an accident since the DSC will still be vacuum dried to remove the air and backfilled with helium before the vent and siphon ports are plugged and welded closed to fully seal the helium filled DSC.

___Yes _X_ No May the consequences of an accident previously evaluated in the SAR be increased?

Since there is no immediate accident scenario associated with the DSC drying operation, the consequences of an accident involving the DSC are not affected by the use of pressurized air or compressed helium for blowdown of the liquid from the DSC enclosure.

ATTACHMENT 3, SAFETY EVALUATION FORM

Page 4 of 5

ACTIVITY: Calvert Cliffs ISFSI USAR Modification 50.59 Log No.____ or 72.48 Log No. 94-0-101-004

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not increased.

___Yes _X_ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

No new malfunctions can be caused by the use of pressurized air in lieu of helium for liquid removal from the DSC. The pressurized air will be supplied by the plant air system. The supplied air will be locally filtered with coalescing filter units rated at 99.9% efficiency to remove extremely small liquid water droplets, oil droplets, and particulates. The maximum oil or hydrocarbon contents of the air will not exceed one part per million for 0.1 micron particulates after filtration. This filtration will provide air quality equal to that used for instrument air. This quality of air is adequate to perform this operation. The insignificant amount of hydrocarbon particulates entering the DSC will be further reduced during the vacuum drying stage. Vacuum drying removes the air from the DSC cavity prior to backfilling it with helium to provide the required inert atmosphere for storage of the fuel. Since the DSC will contain the same final atmosphere required for the long term fuel storage and be sealed in the same manner described previously in the ISFSI USAR, no new malfunctions are created by these changes.

OSS
7/13/94

___Yes _X_ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

No new accidents can be caused by the use of pressurized air in lieu of helium to remove the liquid from the DSC enclosure. The worst accident condition analyzed in the ISFSI USAR occurs when the fuel is stored in a vacuum canister. This condition results in a peak fuel cladding temperature of 393° C which is well below the limit of 570° C. When surrounded by air for a short period of time, the fuel cladding temperature will be well below 393° C. ISFSI-01 (fuel loading procedure) will provide verification sign off steps to ensure that only helium, and not air, is used in the backfilling operation to provide the required inert atmosphere for storage of the fuel.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

___Yes _X_ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
2.2	This section specifies the DSC vacuum steady pressure during canister vacuum drying stage to be less than 3 torr to ensure that all liquid water has evaporated. It also specifies the helium backfill pressure to be 2.5 psig ± 2.5 psi. These pressure limits are not affected by the use of pressurized air in lieu of helium for removal of liquid from the DSC. Vacuum drying and helium backfilling are two operations performed after the liquid removal is completed, and therefore, are not related nor affected by the type of gas used in the liquid removal stage. The margin of safety is therefore not reduced.

ATTACHMENT 3, SAFETY EVALUATION FORM

Page 5 of 5

ACTIVITY: Calvert Cliffs ISFSI USAR Change 50.59 Log No.____ or 72.48 Log No. 94-0-101-004

Summary: (For NRC Report, provide a brief overview)

Complete for 72.48:

___Yes _X_ No Will the proposed activity involve a significant increase in occupational dose?

The use of pressurized air in lieu of helium to force the liquid out of the DSC cavity prior to vacuum drying it and backfilling it with helium does not affect the occupational dose. Table 7.4-1 of Vol. I of the ISFSI USAR gives the estimated dose rates associated with water removal and vacuum drying the DSC cavity (20.8 mrem total personnel dose). This dose rate will not be affected by the above changes.

___Yes _X_ No Will the proposed activity involve a significant unreviewed environmental impact?

The use of pressurized air in lieu of helium for liquid removal from the DSC cavity has no adverse environmental impact nor does it affect the ISFSI Environmental Impact Statement. The Auxiliary Building processing systems are used during the DSC purge and drying operations. During this operation, the liquid and gases (air or helium) purged from the DSC cavity are routed to the Auxiliary Building processing systems or the spent fuel pool.

The ISFSI USAR (Vol. I, Sections 1.3.1.7, 1.3.1.9, 4.3.1, 5.1.1.3) describes the operation of the ISFSI Vacuum Drying System (VDS), which is used to remove water and air from the DSC and replaces it with helium. The system is designed to operate in four modes: liquid removal by pumping, helium forced liquid removal, vacuum pumping, and helium backfilling. This description is changed to allow pressurized air to be used in lieu of helium in the second mode of liquid removal from the DSC cavity. After liquid is forced out by the pressurized air, the DSC will be vacuum dried to remove the air and vapors, and then backfilled with helium to provide the required inert environment for long term fuel clad integrity, as described in the ISFSI USAR. Using air instead of helium to blowdown the water from the DSC cavity, limits the use of helium to the backfilling operation. This results in less use of this gas, and eliminates the presence of it in the atmosphere of the Spent Fuel Area. Helium in the atmosphere could interfere with the function of the closure weld leak detector that is designed to detect helium leakage from the welds of the sealed DSC. The use of pressurized air instead of helium for liquid removal from the DSC cavity does not constitute an unreviewed safety question, a significant increase in occupational exposure nor an unreviewed environmental impact for the Independent Spent Fuel Storage Installation.

CCI-177
Rev. B/Change 0

ATTACHMENT 2, UFSAR CHANGE REQUEST FORM (UCR)

NONMOD # 94-035

To: UFSAR Coordinator
From: Sam Shakir (PRINTED NAME) Work Group CCSO Date 7/8/94
Phone Number: x2179 System Number 101

SECTION 1 *(Change Initiation)*

UFSAR CHANGE SOURCE DOCUMENT

FCR/FEC/MCR # ______ (Circle One)
Safety Evaluation Log # 94-0-101-004
RDC ______
Procedure # ______
License Amendment # ______
Regulatory Generic Correspondence # ______ (Generic Letter, Bulletin or Information Notice)

Unit 1 ____ Unit 2 ____ Common ____ ISFSI X

DESCRIPTION OF UFSAR CHANGE:

1) Change Volume I, Section 1.3.1.7 to read:
"The vacuum drying system removes water and air from the DSC and fills it with helium. The vacuum drying system has four operational modes: water removal, helium or air forced water removal, vacuum pumping, and helium backfilling."

2) Change Volume I, Section 1.3.1.9 item I. to read:
"Air or helium lines are connected to the DSC vent port and the water inside the canister is forced out the siphon tube by pressurized air or helium."

3) Change Volume I, Section 4.3.1 to read:
"The VDS is designed to operate in four modes: liquid removal by pump, liquid removal by a source of pressurized helium or air, vacuum drying, and helium backfill. The evacuation is performed.........still present in the DSC."

4) Change Volume I, Section 5.1.1.3 to read:
"Connect the VDS to the DSC. Open the cask drain port valve and remove the remaining water from the cask/DSC annulus. Remove the remaining water from the DSC cavity by engaging the compressed helium supply or a compressed air source through the helium inlet connection and opening the valve to the DSC vent port, forcing the water from the DSC through the siphon port."

UFSAR SECTIONS AFFECTED: (Attach Marked up Page(s))

ISFSI USAR Volume I, Sections 1.3.1.7, 1.3.1.9, 4.3.1, 5.1.1.3.

ATTACHMENT 2, UFSAR CHANGE REQUEST FORM (UCR)

SECTION 2 (Interdisciplinary Reviews)

RESP. IND. ______________________ WORK GROUP: ______________________
Printed Name and Signature

RESP. IND. ______________________ WORK GROUP: ______________________
Printed Name and Signature

RESP. IND. ______________________ WORK GROUP: ______________________
Printed Name and Signature

SECTION 3 (Implementation Verification Prior to UFSAR Incorporation)

VERIFICATION THAT PLANT MODIFICATION OR AS-BUILT INFORMATION HAS BEEN IMPLEMENTED:

☐ Partial Implementation

(For changes which have been partially implemented, identify the completed portion of the change on the marked-up UFSAR pages. If implementation is complete on one unit only, check the appropriate box, below.)

☐ Unit 1 ☐ Unit 2

RESPONSIBLE ENGINEER: ______________________ DATE: __________

SECTION 4 (Final Review/Approval Prior to UFSAR Incorporation)

FINAL REVIEW & APPROVAL OF THIS CHANGE:

RESPONSIBLE ENGINEER: Jon N. Woodfield Jon N. Woodfield DATE: 7/8/94

RESP. ENGR'S. SUPERVISOR: Michael J. Gahan III DATE: 7.8.94
MICHAEL J. GAHAN, III

UFSAR COORDINATOR: ______________________ DATE: __________

PE-LICENSING UNIT OR WGL: ______________________ DATE: __________

of the DSC to the HSM. Both solid neutron and lead gamma shielding are incorporated into the transfer cask design. Figure 1.3-2 shows the major components of the transfer cask. The Calvert Cliffs transfer cask has a solid hydrogenous neutron shield in the outer annulus of the cask, and as a result the liquid neutron shield expansion tank of Reference 1.2 is deleted.

1.3.1.4 Transfer Trailer [See Reference 1.4]

The transfer trailer is used to transport the transfer cask skid and the loaded transfer cask from the Auxiliary Building to the ISFSI. The transfer trailer is an industrial heavy-haul trailer with pneumatic tires, hydraulic suspension and steering, and brakes on all wheels. Four hydraulic jacks are incorporated into the transfer trailer design to provide vertical elevation adjustment for alignment of the cask at the HSM. The transfer trailer is shown in Figure 1.3-3. It is pulled by a conventional tractor.

1.3.1.5 Transfer Cask Skid and Positioning System

The transfer cask skid is essentially identical in design and operation to previous NUHOMS-24P system transfer cask support skids. The skid is supported on lubricated bearing plates attached to the trailer deck and can be moved horizontally on the bearing plates by the hydraulic actuators of the skid positioning system. The skid is secured to the trailer deck in a travel lock position during cask loading and transport operations. The transfer cask skid is shown in Figure 1.3-4.

1.3.1.6 Hydraulic Ram System

The hydraulic ram consists of a double acting hydraulic cylinder with a capacity of 80,000 lb. in either push or pull and stroke of 21 feet. The ram will be supported during operation by a frame assembly attached to the bottom of the transfer cask and a tripod assembly resting on the concrete slab. The operational loads of the hydraulic ram are grounded through the transfer cask. The hydraulic ram system includes a grapple at the end of the piston which is used to engage a grapple ring on the DSC for retrieval operations. Figure 1.3-5 shows the hydraulic ram system.

1.3.1.7 Vacuum Drying System

The vacuum drying system removes water and air from the DSC and fills it with helium. The vacuum drying system has four operational modes: water removal, helium or air forced water removal, vacuum pumping, and helium backfilling.

1.3.1.8 Automated Closure Welding System

The DSC closure welds on the shield plug and the top cover plate are placed by a fully remote, automatic welding system. The system includes modular components and is designed for rapid setup. Welding operations are remotely controlled by an operator who views the progress of the weld through closed circuit television. The welding head is designed to permit rapid replacement with either a UT probe, or a plasma gouging torch which can be used to remove the shield plug and top cover plate closure welds.

1.3.1.9 System Operation

The primary operations, in sequence of occurrence, for the Calvert Cliffs system are shown schematically in Figure 1.3-6 and are described below:

A. Transfer Cask Preparation - Cask preparation includes exterior washdown and interior decontamination if necessary.

B. DSC Preparation - The canisters are thoroughly cleaned.

C. DSC/Transfer Cask Loading - The empty DSC is inserted into the transfer cask using the Spent Fuel Cask Handling Crane and lifting lugs provided on the DSC. Proper angular alignment is achieved through the use of alignment marks on the cask and each DSC.

D. Transfer Cask Lifting and Placement in the Spent Fuel Pool - The annulus between the DSC and cask is filled with demineralized water and sealed with an inflatable seal to prevent contamination of the DSC outer surface by the pool water. Prior to placing the cask in the spent fuel pool, the DSC is filled with fuel pool water to prevent an inrush of water when the cask is lowered into the pool. The cask and DSC are then lowered into the pool.

E. DSC Fuel Loading - Twenty-four spent fuel assemblies are loaded into the DSC basket. These assemblies will be preselected to control reactivity and decay heat using the administrative controls on burnup, initial enrichment, and post-irradiation decay time as detailed in Section 10.2.5.

F. DSC Shield Plug Placement - With the transfer cask and loaded DSC resting in the fuel pool, the DSC shield plug is lowered into place using the Spent Fuel Cask Handling Crane.

G. Transfer Cask Lifting Out of the Pool - The transfer cask and loaded DSC are lifted out of the spent fuel pool and placed in the cask washdown pit using the Spent Fuel Cask Handling Crane. The transfer cask and DSC cover are then decontaminated.

H. DSC Sealing - Initially the water level in the DSC/transfer cask annulus is lowered approximately 5-10 inches. The inflatable seal is removed and swipes are taken over the DSC exterior at the DSC upper surface and around the circumference. The water level in the DSC is lowered to just below the inner surface of the shield plug and a seal weld is made between the shield plug and the DSC shell. This weldment provides the primary closure for the DSC.

I. Transfer Cask/DSC Drying - Air or Helium lines are connected to the DSC vent port and the water inside the canister is forced out the siphon tube by pressurized air or helium. The water in the transfer cask annulus is also drained. The water is returned to the spent fuel pool or routed to Auxiliary Building processing systems. The DSC vent line is then used to draw a vacuum to facilitate drying until the DSC moisture content meets the applicable limits.

J. Helium Filling - In order to ensure that no fuel and/or cladding oxidation occurs during storage, the DSC is filled with helium after evacuation.

4.3 AUXILIARY SYSTEMS

The ISFSI is a self-contained, passive storage facility which requires no auxiliary systems.

4.3.1 VENTILATION AND OFF-GAS SYSTEMS

Spent fuel confined in storage at the ISFSI is cooled by conduction and radiation within the DSC, and conduction, convection, and radiation from the DSC surface. An air inlet near the bottom of the HSM front wall and outlets in the HSM roof allow convective cooling by natural circulation. The driving force for this ventilation system is described in Section 8.1.3. No auxiliary ventilation is used or required at the ISFSI. Fuel loading and DSC closure operations take place in the plant's Auxiliary Building and make use of the ventilation system in that facility. Auxiliary Building ventilation is discussed in Section 9.8.2.3 of Reference 4.2.

The Vacuum Drying System (VDS) provides a means for removing water and water vapor from the DSC and for backfilling the DSC with helium. This function is required to ensure that fuel is stored in an inert atmosphere, and to take advantage of the favorable heat transfer properties of helium.

removal by a source of pressurized helium or air

The VDS is designed to operate in four modes: liquid removal by pump, liquid removal by helium pressure, vacuum drying, and helium backfill. The evacuation is performed in several stages to allow the DSC pressure to stabilize. When the pressure can be held at 3 torr for at least 30 minutes, the cavity is then backfilled with helium. After again pumping the cavity down to 3 torr, a final helium backfill is made and the DSC is sealed. This process further reduces the partial pressure of any water vapor still present in the DSC.

4.3.2 ELECTRICAL SYSTEMS

No electrical systems are required for the HSM or DSC during long term storage, other than for lighting and security system power. Electrical power is used during DSC closure operations in the plant's Auxiliary Building and during DSC transfer operations to the HSM at the ISFSI. The required electrical power in the Auxiliary Building will be obtained from the existing plant system. Power at the ISFSI will be supplied from a retail source.

4.3.3 AIR SUPPLY SYSTEMS

No air supply system is required. Compressed helium will be used to force water from the DSC during closure operations.

4.3.4 STEAM SUPPLY AND DISTRIBUTION SYSTEM

There are no steam systems required.

4.3.5 WATER SUPPLY SYSTEM

Borated water will be used to fill the DSC cavity prior to insertion into the spent fuel pool. The water source will be compatible with the plant's existing spent fuel pool. The source of supply may be the pool itself. Demineralized water is needed for filling the DSC/cask annulus, and for washdown operations.

one of the assemblies selected for storage from the fuel rack and position it over the DSC. Insert the assembly into the basket guide sleeve according to the DSC loading plan and repeat until all guide sleeves are filled. After the DSC has been fully loaded, check and record the identity and location of each fuel assembly in the DSC using an underwater TV camera or special optical equipment suitable for this purpose. When the identity of all fuel assemblies in the DSC has been verified, position the shield plug assembly over the DSC, and lower it until it is properly seated.

Engage the lifting yoke to the cask trunnions and verify visually that it is properly positioned and engaged. Raise the transfer cask to the pool surface, stopping vertical movement prior to breaking the surface of the pool. Inspect the top shield plug to verify that it is properly seated on the DSC. If it is not, lower the cask and reposition the shield plug assembly. Raise the cask from the pool while spraying the exposed portion with demineralized water. Drain any excess water from the top of the DSC shield plug assembly back into the pool. Check the radiation levels at the center and perimeter of the top shield plug assembly and around the exposed surface of the cask. Lift the cask from the pool and move it to the cask washdown pit.

5.1.1.3 Cask/DSC Drying Process [See Reference 5.2]

Disengage the rigging cables from the top shield plug and remove the eyebolts. Disengage the lifting yoke from the trunnions and move it clear of the cask. Check the radiation levels along the surface of the cask and decontaminate it as necessary. Place scaffolding around the cask so that any point on its surface is easily accessible to personnel. Decontaminate the top shield plug surface and the exposed DSC shell, and remove the inflatable cask/DSC annulus seal. Connect the cask drain line to the cask, open the cask cavity drain port, and allow water to drain from the annulus until the water level is approximately twelve inches below the top edge of the DSC shell. Take swipes around the outer surface of the DSC shell and check for removable contamination. Dry the top shield plug surface and exposed interior of the DSC shell above the top lead plug. Check radiation levels along the surface of the top shield plug and install temporary shielding as necessary to minimize personnel exposure.

Connect the vacuum drying system (VDS) to the DSC siphon and vent ports, and use the liquid pump to pump approximately 60 gallons of water from the canister to the fuel pool in order to lower the water level in the DSC below the vent port opening. Disconnect the VDS from the DSC, and install a short stub tube to the vent port fitting to ensure that the DSC internal pressure remains atmospheric during the closure weld operation. Install the automatic welding machine and tack weld the top shield plug to the DSC shell. Place the shield plug seal weldment and remove the automatic welding machine.

the compressed helium supply or a compressed air source through the helium inlet connection and

Connect the VDS to the DSC. Open the cask drain port valve and remove the remaining water from the cask/DSC annulus. Remove the remaining water from the DSC cavity by engaging the compressed helium supply and opening the valve to the DSC vent port, forcing the water from the DSC through the siphon port. When water stops flowing from the DSC, close the siphon port valve. Open the valve on the suction side of the vacuum pump, start the pump, and draw a vacuum of 3 torr or less in the DSC cavity. The pressure in the DSC should be reduced in steps to prevent the formation of ice in the DSC cavity or in the VDS. After pumping down to each level, the pump should be valved off and the cavity pressure monitored. The cavity pressure will rise as water and other volatiles in the

VECTRA

July 11, 1994
BGE01-94-1028

Mr. Robert H. Beall
Baltimore Gas & Electric Company
Calvert Cliffs Nuclear Power Plant
Lusby, MD 20657

Subject: Calvert Cliffs NUHOMS® ISFSI Project - Additional Information to Support Use of Air or Helium for Initial Draining of the DSC after Fuel Load

Dear Mr. Beall:

In a telephone conversation between BG&E (Bob Beall) and VECTRA (M. Taylor), BG&E requested the following information regarding the use of air or helium for initial draining of the DSC after fuel load:

1. Are there any restrictions on the quality of the air used for the draining? Is normal plant air acceptable?
2. Is there a time limit on how long the canister internals and fuel can be exposed to the air environment?

VECTRA's responses to the above questions are as follows:

1. Normal plant air is acceptable for the DSC initial draindown.
2. The initial draindown operation is followed immediately by the evacuation and helium backfilling operation. This limits the time that the canister internals and fuel are exposed to an air environment to approximately eight hours. Thermal calculations show that the short term (up to several weeks) fuel cladding temperature limits are not exceeded in a vacuum environment. Since an air environment is less severe than a vacuum environment from a thermal standpoint, short term exposure to air is acceptable from a thermal standpoint. Also, an air environment is no more corrosive to the exposed materials than water in the short term.

VECTRA Technologies, Inc. · 6203 San Ignacio Ave., Suite 100 · San Jose, CA 95119 · Tel: (408) 629-9800
Fax: (408) 281-6166 Engineering · Fax: (408) 281-6202 Fuel Services

If you have any additional questions, please contact me.

Sincerely,

Moses Taylor Jr

Moses Taylor, Jr., P.E.
Project Manager

cc: P. A. File
J. B. Makar

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 1

This screening is for: ______**10 CFR 50.59 Applicability** __x__ **10 CFR 72.48 Applicability**
(Check one regulation only)

______**CCNPP** __x__**ISFSI**
(Check one facility only)

(Check one activity type only)

______**Procedure:** Procedure No./Change No.:__________________

______**Temporary Alteration:** Temporary Alteration No.:__________________

______**Setpoint Change:** SCAF No(s):__________________

______**Modification:** MCR/FCR/FEC No.:__________________
FEC Supplement No.:__________________

______**Core Reload:** Unit and Cycle:__________________

__x__**UFSAR/USAR:** UFSAR/USAR Change No.: 94-035

______**Other:** Identify Activity Type:__________________

Brief description of the activity:

Change the ISFSI USAR, Vol. I, Sections 1.3.1.7, 1.3.1.9, 4.3.1, and 5.1.1.3 to allow the use of pressurized air or helium for liquid removal from the ISFSI Dry Shielded Canister (DSC) by the Vacuum Drying System (VDS). The current ISFSI USAR describes the VDS operation in four modes: liquid removal by pumping, helium forced liquid removal, vacuum pumping, and helium backfilling. The change only affects the second mode of the VDS operation, where the use of pressurized air or helium is allowed for forced liquid removal from the DSC cavity. The benefits of using air instead of helium is to save a significant amount of helium needed for the blowdown of liquid from the DSC, and to eliminate the presence of helium concentration in the atmosphere of the Spent Fuel Pool area which could interfere with the function of the helium leak detector used for measuring leakage from the DSC inner cover plate closure weld. The presence of helium in the air could result in a delay of the DSC closure operations until the helium concentration is removed by the Auxiliary Building ventilation system.

Technical Specifications/License Conditions (10 CFR 50.59/72.48)

1.____YES __x__NO Is the proposed activity a change or will it cause a change to the Technical Specifications/License Conditions or Bases?

2.____YES __x__NO Will the proposed activity cause Structures, Systems or Components (SSCs) to be operated in a manner that violates the Technical Specifications/License Conditions or Bases?

If both answers are **"No,"** continue with the screening. Justification for each **"No"** answer shall be provided. List the sections of the Technical Specifications/License Conditions that were reviewed.

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 2

Justification:

The change to the ISFSI USAR description of the VDS operation to allow the use of air or helium in the liquid removal mode does not impact any technical specification. After liquid removal is complete, the DSC cavity will be vacuum dried and backfilled with helium as specified in the ISFSI Technical Specification.

Technical Specifications/License Condition Sections Reviewed:

Reviewed ISFSI Technical Specification, Section 2.2.

If either of the above answers is **"Yes,"** complete a Safety Evaluation and consult CCI-143 for License Amendment Proposals.

CCNPP/ISFSI Facility (10 CFR 50.59/72.48)

1.____YES __x__NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of the structure, system or component (SSC) directly affected by the activity?

If "No," answer each question below:

Why is the SAR description of the function of the SSC not affected?

The DSC provides containment and confinement of the spent fuel during storage. Using pressurized air instead of helium for liquid removal from the DSC cavity during the drying operation does not affect the containment and confinement function of the DSC. The VDS provides a means for removing water and air from the DSC cavity and for backfilling the DSC with helium. The use of air instead of helium in the second stage of the VDS operation to force water out of the DSC cavity has no affect on the function of the VDS. The DSC will still be vacuum dried to remove the air and vapors and then backfilled with helium and sealed as described in the ISFSI USAR.

Why is the SAR description of the method of performing the function of the SSC not affected?

The drying function of the VDS is performed by using pressurized gas to force the liquid out of the DSC cavity. There is no change in the method of performing the drying function of the VDS whether air or helium is pumped into the DSC cavity. Therefore, the use of pressurized air is acceptable and does not affect the method of performing the function of either the VDS or the DSC.

Why is the SAR description of the design of the SSC not changed?

The VDS is designed to remove water and air from the DSC and to backfill the DSC with helium. The VDS is designed to operate in four modes: liquid removal by pumping, forced liquid removal by pressurized gas, vacuum pumping, and helium backfilling. Permitting the use of air instead of helium in the second stage of this operation to force the liquids out of the DSC cavity has no affect on the design of the VDS or the DSC. The atmospheric environment inside the DSC cavity required for the long term dry fuel storage is not affected by this change. The DSC will still be vacuum dried and backfilled with helium, as described in the ISFSI USAR, to provide the required inert environment for long term fuel clad integrity.

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 3

2.____YES __x__NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of any other SSC described in the SAR?

If "No," answer the following question:

Explain why the activity does not affect other SSCs described in the SAR..

No other SSCs are affected by this activity. The final sealed inert environment required for long term storage of the spent fuel inside the DSC cavity is not affected by this change.

3.__x__YES ______NO Is the proposed activity a revision to the SAR. (Editorial changes are limited to obvious grammatical/spelling errors, reorganization of portions of the SAR or minor changes that do not affect the intent of the information conveyed by a drawing.)

4.__x__YES ______NO Will the proposed activity add to or delete from the SAR description of a SSC?

Procedures (10 CFR 50.59/72.48)

1.____YES __x__NO Will the proposed activity affect the intent of any procedure described in the SAR (editorial changes do not need a Safety Evaluation)? The NRC staff does not consider procedures simply listed in the SAR to be described in the SAR. Also, procedures include anything that defines or describes activities or controls over functions, tasks, reviews, tests and safety review meetings.

2.____YES __x__NO Will the proposed activity cause SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: The activity allows the use of pressurized air or helium for liquid removal from the DSC cavity during the drying operation of the DSC using the VDS. This change does not affect any procedures outlined in the ISFSI USAR. The VDS four mode operation will not change, nor will the final inert environment inside the DSC. Therefore, the change does not impact the design, function, or method of performing the function of the DSC, or VDS.

Tests or Experiments (10 CFR 50.59/72.48)

1.____YES __x__NO Will the proposed activity result in conducting a test or experiment causing SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: This activity is not a test or experiment.

ISFSI (10 CFR 72.48) These questions are only required to be answered for activities affecting ISFSI.

1.____YES ___x___NO Will the proposed activity increase any occupational dose for ISFSI related activities?

2.____YES ___x___NO Will the proposed activity use additional property for ISFSI operations?

3.____YES ___x___NO Will the proposed activity add or change the roads or transport equipment, including cranes, used for ISFSI operations?

Justify each "No" answer below:

Justification: This activity allows the use of air or helium for liquid removal from the DSC cavity during the drying operation. The liquid removal and drying operation using the VDS remains unchanged with no impact to the occupational dose associated with it. The drying activity takes place in the Cask Wash Pit on the 69' level of the Auxiliary Building. No additional ISFSI property nor changes to road transport or equipment is required or included in this activity.

SAR Sections Reviewed:

Volumes I, IV, & V of the ISFSI USAR

If **ALL** answers are "No", A Safety Evaluation is not required.

If **ANY** answer is "Yes", A Safety Evaluation is required.

1.___x___YES ______NO Does this activity require additional screening?

✓ **10CFR 50.59** For Impact on CCNPP
10 CFR 72.48 For Impact on ISFSI

If "Yes", Perform a separate Safety Evaluation Screening.

Prepared By: Sam Shakir Sam Shakir Date: 7/8/14
PRINTED NAME AND SIGNATURE

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 1

This screening is for: __x__ **10 CFR 50.59 Applicability** ____ **10 CFR 72.48 Applicability**
(Check one regulation only)

__x__ **CCNPP** ____ **ISFSI**
(Check one facility only)

(Check one activity type only)

____ **Procedure:**	Procedure No./Change No.:
____ **Temporary Alteration:**	Temporary Alteration No.:
____ **Setpoint Change:**	SCAF No(s):
____ **Modification:**	MCR/FCR/FEC No.: FEC Supplement No.:
____ **Core Reload:**	Unit and Cycle:
__x__ **UFSAR/USAR:**	UFSAR/USAR Change No.: 94-035
____ **Other:**	Identify Activity Type:

Brief description of the activity:

Change the ISFSI USAR, Vol. I, Sections 1.3.1.7, 1.3.1.9, 4.3.1, and 5.1.1.3 to allow the use of pressurized air or helium for liquid removal from the ISFSI Dry Shielded Canister (DSC) by the Vacuum Drying System (VDS). The current ISFSI USAR describes the VDS operation in four modes: liquid removal by pumping, helium forced liquid removal, vacuum pumping, and helium backfilling. The change only affects the second mode of the VDS operation, where the use of pressurized air or helium is allowed for forced liquid removal from the DSC cavity. The benefits of using air instead of helium is to save a significant amount of helium needed for the blowdown of liquid from the DSC, and to eliminate the presence of helium concentration in the atmosphere of the Spent Fuel Pool area which could interfere with the function of the helium leak detector used for measuring leakage from the DSC inner cover plate closure weld. The presence of helium in the air could result in a delay of the DSC closure operations until the helium concentration is removed by the Auxiliary Building ventilation system. Only ISFSI SSCs are affected by this change, however, this screen is required since the activity takes place on the 69' level of the Auxiliary Building which is a 10CFR 50.59 territory. No other SSCs inside the Auxiliary Building are affected by this change.

Technical Specifications/License Conditions (10 CFR 50.59/72.48)

1. ____ YES __x__ NO Is the proposed activity a change or will it cause a change to the Technical Specifications/License Conditions or Bases?

2. ____ YES __x__ NO Will the proposed activity cause Structures, Systems or Components (SSCs) to be operated in a manner that violates the Technical Specifications/License Conditions or Bases?

If both answers are **"No,"** continue with the screening. Justification for each **"No"** answer shall be provided. List the sections of the Technical Specifications/License Conditions that were reviewed.

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 2

Justification:

A description of the VDS drying operation appears only in the ISFSI USAR and the ISFSI Technical Specifications. No such description appears in the UFSAR or the Plant Technical Specifications.

Technical Specifications/License Condition Sections Reviewed:

Reviewed all sections of the CCNPP Technical Specifications. None are applicable to this activity.

If either of the above answers is **"Yes,"** complete a Safety Evaluation and consult CCI-143 for License Amendment Proposals.

CCNPP/ISFSI Facility (10 CFR 50.59/72.48)

1.____YES ___x__NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of the structure, system or component (SSC) directly affected by the activity?

If "No," answer each question below:

Why is the SAR description of the function of the SSC not affected?

The activity has no impact on the function of the DSC and VDS as described in the ISFSI USAR. No SSCs described in the UFSAR are affected by the liquid removal operation from the DSC. The liquids and gases removed from the DSC will still be routed to the Auxiliary Building Processing System or the Spent Fuel Pool as described in the ISFSI USAR. Therefore, this activity does not affect the function of any SSCs in the Auxiliary Building.

Why is the SAR description of the method of performing the function of the SSC not affected?

The drying operation of the DSC, which takes place in the Auxiliary Building, will remain unchanged by the use of pressurized air instead of helium for liquid removal from the DSC cavity. No SSCs described in the UFSAR are affected by this change.

Why is the SAR description of the design of the SSC not changed?

No SSCs described in the UFSAR are affected by this change.

2.____YES ___x__NO Will the proposed activity result in a change to the SAR description of the design, function or method of performing the function of any other SSC described in the SAR?

If "No," answer the following question:

Explain why the activity does not affect other SSCs described in the SAR..

This is an ISFSI activity involving ISFSI components only that takes place inside the Auxiliary Building. No other SSCs described in the UFSAR are affected by this change.

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 3

3.____YES ___x___NO Is the proposed activity a revision to the SAR. (Editorial changes are limited to obvious grammatical/spelling errors, reorganization of portions of the SAR or minor changes that do not affect the intent of the information conveyed by a drawing.)

4.____YES ___x___NO Will the proposed activity add to or delete from the SAR description of a SSC?

Procedures (10 CFR 50.59/72.48)

1.____YES ___x___NO Will the proposed activity affect the intent of any procedure described in the SAR (editorial changes do not need a Safety Evaluation)? The NRC staff does not consider procedures simply listed in the SAR to be described in the SAR. Also, procedures include anything that defines or describes activities or controls over functions, tasks, reviews, tests and safety review meetings.

2.____YES ___x___NO Will the proposed activity cause SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: The activity allows the use of pressurized air in place of helium for liquid removal from the DSC cavity during the drying operation of the DSC. This is an ISFSI activity that takes place inside the Auxiliary building. This change does not affect any procedures outlined in the UFSAR, nor does it impact the design, function, or method of performing the function of any SSCs described in the UFSAR.

Tests or Experiments (10 CFR 50.59/72.48)

1.____YES ___x___NO Will the proposed activity result in conducting a test or experiment causing SSCs to be operated in a manner that is not consistent with the design, function, or method of performing the function, as described in the SAR?

Justify each "No" answer below:

Justification: This activity is not a test or experiment.

ISFSI (10 CFR 72.48) These questions are only required to be answered for activities affecting ISFSI.

1.____YES ______NO Will the proposed activity increase any occupational dose for ISFSI related activities?

2.____YES ______NO Will the proposed activity use additional property for ISFSI operations?

ATTACHMENT 2, SAFETY EVALUATION SCREENING FORM
Page 4

3._____YES ______NO Will the proposed activity add or change the roads or transport equipment, including cranes, used for ISFSI operations?

Justify each "No" answer below:

Justification:

SAR Sections Reviewed:

Volumes I, IV, & V of the ISFSI USAR

If **ALL** answers are "No", A Safety Evaluation is not required.

If **ANY** answer is "Yes", A Safety Evaluation is required.

1.__x__YES ______NO Does this activity require additional screening?

10CFR 50.59 For Impact on CCNPP
✓ **10 CFR 72.48** For Impact on ISFSI

If "Yes", Perform a separate Safety Evaluation Screening.

Prepared By: Sam Shakir Sam Shakir Date: 7/8/94
PRINTED NAME AND SIGNATURE

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 1)

ACTIVITY: MCR 93-031-003-01 50.59 Log No.: N/A 72.48 Log No.: 94-B-0312-005-R00

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

___YES X__NO Involve an unreviewed safety question (USQ)?

___YES X__NO Involve a change in the Technical Specifications/License Conditions or Bases?

X__YES ___NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

___YES X__NO Involve a Significant Increase in Occupational Dose?

___YES X__NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Kirk A. Kondos Department: PDSU Date: 11/30/94

PRINTED NAME AND SIGNATURE

___YES X__NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.:__________ Resp. Ind.:__________ Resp. Ind.:__________

PRINTED NAME PRINTED NAME PRINTED NAME

SIGNATURE SIGNATURE SIGNATURE

Work Group:__________ Work Group:__________ Work Group:__________

Date:__________ Date:__________ Date:__________

Approved X Disapproved _ Approved X Disapproved __

Signature__________ Signature__________

INDEPENDENT REVIEWER GS-DES,GS-TSES, OR PE-PDSU

Date: 12/7/94 Date: 12/7/94

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 2)

ACTIVITY: MCR 93-031-003-01 50.59 Log No.: N/A 72.48 Log No.: 94-B-0312-005-R00

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 94-168 Date: 12/14/94

Recommend Approval ✓ Recommend Disapproval ____ Signature ____ Date 12-14-94

POSRC CHAIRMAN

Approved ✓ Disapproved ____ Signature FOR CHC Date 12/14/94

PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

OSSRC Meeting No.: 96-005 Date: 5/16/96

Recommend Approval ____ Recommend Disapproval ____ Signature ____ Date ____

OSSRC CHAIRMAN

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 3)

ACTIVITY: MCR 93-031-003-01 50.59 Log No.: N/A 72.48 Log No.: 94-B-0312-005-R00

Proposed Activity:

The proposed activity retires the backup meteorological instruments located on the microwave tower as described in USAR Section 2.3.3, On-Site Meteorological Measurement Program, Figure 2.3-2 (Meteorological Instrument Elevations), Figure 2.3-3 (Meteorological Data Acquisition System) and Table 2.3-2 (On-Site Meteorological Stations and Instrumentation). This USAR Section will be revised by this proposed activity by removing all references to the backup meteorological instruments located on the microwave tower or stating they are spare.

Reason of Activity:

The backup meteorological instruments located on the microwave tower are old and use obsolete equipment. This equipment requires a significant amount of maintenance to remain operational. The backup meteorological system is of such design that it creates a detrimental maintenance environment for technicians replacing and repairing equipment.

Function(s) of affected SSC:

The function of the backup meteorological instruments located on the microwave tower was to provide meteorological information to the control room for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment. Information is displayed to the control room on the plant computer and the technical support center (MIDAS) computer. This function of the backup meteorological instruments located on the microwave tower will be eliminated by this activity.

The plant computer function is to assist the control room operators in the safe and efficient operation of each unit. This activity simply removes inputs from the backup meteorological instruments located on the microwave tower and the switchyard building to the plant computer. The inputs from the backup meteorological instruments located on the microwave tower and the switchyard building are not used by the control room operators in the safe and efficient operation of each unit.

The function of the Technical Support Center Computer is to provide selected plant status information to support staff assigned to the TSC during designed times. This information is available on display monitors (MIDAS), printers and trend recorders. The TSC computer enables the support staff to monitor and assess the status of the plant and assist the control room operators in analyzing events and safely stabilizing the plant. The inputs from the backup meteorological instruments located on the microwave tower and the switchyard building to the TSC have been duplicated by inputs from the meteorological tower.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 4)

ACTIVITY: MCR 93-031-003-01 50.59 Log No.: N/A 72.48 Log No.: 94-B-0312-005-R00

SAR Sections Reviewed:

USAR Section 2.3.3, On-Site Meteorological Measurement Program, Figure 2.3-2 (Meteorological Instrument Elevations), Figure 2.3-3 (Meteorological Data Acquisition System) and Table 2.3-2 (On-Site Meteorological Stations and Instrumentation) was reviewed. This USAR Section will be revised by this proposed activity by removing all references to the backup meteorological instruments located on the microwave tower or stating they are spare.

Technical Specification 3/4.3.3 provides requirements for Technical Specification-related meteorological instrumentation. Table 3.3-8 lists the required meteorological monitoring instrumentation channels. All of the instrumentation listed on this table is mounted on the primary tower. None of the instrumentation on the backup meteorological tower is required by the Technical Specifications.

NUREG-0654 requires each site to have a viable backup meteorological system to provide meteorological information when the primary system is out of service. The acceptance criteria for the backup meteorological system are described in the proposed Revision 1 to Regulatory Guide 1.23. Regulator Position C.8 of Regulatory Guide 1.23, Revision 1 recommends that an independent system or procedure be established for obtaining measurements of wind direction and speed representative of the 10-meter level and an estimate of the atmospheric stability (e.g., temperature difference with height, wind direction fluctuations). It is important to note that the backup tower is described in Regulatory Position (8) ONLY, and is not required to meet the other seven criteria in the Regulatory Position section of this Regulatory Guide. Additionally, the backup meteorological instruments on the microwave tower satisfy the requirements of Regulatory Guide 1.23, Revision 1, for an independent system, as described in letter from Mr. A. E. Lundvall, Jr. (BG&E) to MR. T. T. Martin (NRC), dated February 8, 1985, "Radiological Dose Assessment Capability During Emergencies".

In addition to the regulatory guidance described above, Regulatory Guide 1.97, Revision 3 specifies additional requirements for meteorological instrumentation. Meteorological assessment is considered a Category 3 variable. However, redundancy is not required for Category 3 instrumentation; therefore, the backup meteorological tower is not required to meet the requirements of this Regulatory Guide. Letter from J. A. Tiernan (BG&E) to NRC Document Control Desk, dated August 9, 1988, "Regulatory Guide 1.97 Review and Update" describes how Calvert Cliffs' primary meteorological tower meets the requirements of Regulatory Guide 1.97.

Calvert Cliffs had implemented both an independent procedure and system using the back up tower for obtaining meteorological information. ERPIP 825, Revision 0 provided instructions for obtaining wind speed and direction data from Patuxent River Naval Air Station, and for determining atmospheric stability from outside observation, if both the primary and backup meteorological instrumentation is nonfunctional. A 10 CFR 50.54(q) (POSRC approved on November 1, 1993) has revised ERPIP to Revision 1 which no longer references the backup meteorological instrumentation. This 10 CFR 50.54(q) has also revised ERP Revision 17, Section 5.III.A., Geophysical Phenomena Monitors, deleted the reference to a backup tower in lieu of reference to the Emergency Response Plan Implementation Procedures which provides a backup method for obtaining meteorological data.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 5)

ACTIVITY: MCR 93-031-003-01 50.59 Log No.: N/A 72.48 Log No.: 94-B-0312-005-R00

Complete for 50.59 and 72.48:

1. **The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.**

___ Yes X__ No May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The postulated malfunction is a malfunction of the backup meteorological system.

The wording of NUREG 0654 and Reg. Guide 1.23 allows independent systems OR procedures to be established as backup methods (for obtaining measurements of wind direction, wind speed and an estimate of atmospheric stability). Calvert Cliffs Emergency Response Plan Implementation Procedures have established a backup method for obtaining wind speed and direction from Patuxent River Naval Air Station. Backup atmospheric stability estimates are derived from sigma theta instruments (on the primary meteorological tower), and a method for determining atmospheric stability from outside observation if measurements are unavailable. These procedures meet the requirements of NUREG 0654 and Reg. Guide 1.23. Since these independent methods are adequate to provide required backup, deletion of the backup meteorological instruments located on the microwave tower does not increase the probability of malfunction of equipment important to safety as previously evaluated in the SAR.

___ Yes X__ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The radiological consequences have not increased. This activity removes data inputs from the backup meteorological instruments located on the microwave tower to the Technical Support Center Computer and the plant computer via the DAS. The meteorological tower currently is a data input to the Technical Support Center Computer. The removal of the data inputs from the backup meteorological instruments will not change the anticipated plant response to any malfunction. Therefore, the consequences of a malfunction of equipment important to safety are not increased.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 6)

ACTIVITY: MCR 93-031-003-01 50.59 Log No.: N/A 72.48 Log No.: 94-B-0312-005-R00

___ Yes X__ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

None of the equipment associated with the backup meteorological instruments located on the microwave tower represents an accident initiator, therefore there is no increase in the probability of an accident.

___ Yes X__ No May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The function of the Technical Support Center Computer and the plant computer is unaffected by the removal of the data inputs from the backup meteorological instruments located on the microwave tower. The backup meteorological instruments located on the microwave tower are not credited and play no role in the accident mitigation. Revision 1 to the ERPIP no longer references the backup meteorological instrumentation. ERP Revision 17, Section 5.III.A., Geophysical Phenomena Monitors, deleted the reference to a backup tower. Therefore, any assumptions made in evaluating the radiological off-site dose to the public are not altered. Therefore, the consequences of any accident previously evaluated in the SAR are not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

___ Yes X__ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

As stated in paragraph i.A, the Calvert Cliffs Emergency Response Plan Implementation Procedures have established a backup method for obtaining wind speed, wind direction and atmospheric stability. These procedures meet the requirements of NUREG 0654 and Reg. Guide 1.23. Since these independent methods are adequate to provide required backup, deletion of the backup meteorological instruments located on the microwave tower does not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 7)

ACTIVITY: MCR 93-031-003-01 50.59 Log No.: N/A 72.48 Log No.: 94-B-0312-005-R00

___ Yes X__ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

This activity does not create or increase the possibility of an accident. The backup meteorological instruments located on the microwave tower are passive devices that only provide control room indication. Therefore, this activity does not create or increase the possibility of an accident during any mode.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

___ Yes X__ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
3/4.3.3	Technical Specification 3/4.3.3 provides requirements for Technical Specification-related meteorological instrumentation. Table 3.3-8 lists the required meteorological monitoring instrumentation channels. All of the instrumentation listed on this table is mounted on the primary tower. None of the instrumentation on the backup meteorological tower is required by the Technical Specifications.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 8)

ACTIVITY: MCR 93-031-003-01 **50.59 Log No.: N/A 72.48 Log No.: 94-B-0312-005-R00**

Complete for 72.48:

___ Yes X__ No Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

This activity does not have any affect on Occupational Dose. The backup meteorological instruments located on the microwave tower are a passive device that only provides control room indication.

___ Yes X__ No Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

This activity does not affect any area of the plant site previously undisturbed for the ISFSI installation. This activity does not revise the ISFSI Environmental Impact Statement. The backup meteorological instruments are located on the microwave tower in the switchyard.

Summary: (For NRC Report, provide a brief overview)

The proposed activity retires the backup meteorological instruments located on the microwave tower as described in USAR Section 2.3.3, On-Site Meteorological Measurement Program, Figure 2.3-2 (Meteorological Instrument Elevations), Figure 2.3-3 (Meteorological Data Acquisition System) and Table 2.3-2 (On-Site Meteorological Stations and Instrumentation). This USAR Section will be revised by this proposed activity by removing all references to the backup meteorological instruments located on the microwave tower or stating they are spare.

This activity does not constitute an Unreviewed Safety Question (USQ). This activity has no affect in the occupational dose and does not involve a significant unreviewed environmental impact for the ISFSI installation.

Calvert Cliffs Emergency Response Plan Implementation Procedures have established a backup method for obtaining wind speed and direction from Patuxent River Naval Air Station. Backup atmospheric stability estimates are derived from sigma theta instruments (on the primary meteorological tower), and a method for determining atmospheric stability from outside observation if measurements are unavailable. These procedures meet the requirements of NUREG 0654 and Reg. Guide 1.23. Since these independent methods are adequate to provide required backup, deletion of the backup meteorological instruments located on the microwave tower from the Emergency Response Plan does not reduce the plan's effectiveness.

ATTACHMENT 3, SAFETY EVALUATION FORM (Page 1 of 4)

Page 1 **of** 4

ACTIVITY: ISFSI USAR CHANGE **50.59 Log No.:** NA **72.48 Log No.:** 95-0001 SE 00001

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

__ YES X NO Involve an unreviewed safety question (USQ)?
__ YES X NO Involve a change in the Technical Specifications/License Conditions or Bases?
X YES __ NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

__ YES X NO Involve a Significant Increase in Occupational Dose?
__ YES X NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: M.A. CARR Department: NED/DES/CEU Date: 6/30/95
PRINTED NAME AND SIGNATURE
Jon N. Woodfield Jon N. Woodfield 7/5/95

X YES __ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: [illegible] PRINTED NAME	Resp. Ind.: R.H. Beall PRINTED NAME	Resp. Ind.: G. Tesfaye PRINTED NAME
SIGNATURE	R.H. Beall SIGNATURE	SIGNATURE
Work Group: PES	Work Group: NFM	Work Group: LICENSING
Date: 7/6/95	Date: 7/10/95	Date: 7-5-95

Approved ✓ Disapproved __	Approved ✓ Disapproved __
Signature G.V. Patel G.V. PATEL INDEPENDENT REVIEWER 7-15-95	Signature Michael J. Gahan for GS-DES, GS-TSES, or PE-PDSU
Date 7-15-1995	Date 8.17.95

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 95-97 Date: 8-23-95

Recommend Approval ✓ Recommend Disapproval ____ Signature ____ Date 8-23-95
POSRC CHAIRMAN

Approved ✓ Disapproved ____ Signature ____ Date 8/24/95
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
OSSRC Meeting No.: 95-001 Date: 11-22-95

Recommend Approval ____ Recommend Disapproval ____ Signature ____ Date ____
OSSRC CHAIRMAN

ATTACHMENT 3, SAFETY EVALUATION FORM

(Page 2 of 4)

SE00001

ACTIVITY: Calvert Cliffs ISFSI USAR Change **50.59 Log No:** NA **72.48 Log No:** 95-0001

Proposed Activity: *Upgrade the site's vehicle barrier systems to prevent access by a malevolent vehicle within the Safe Standoff Distance from selected CCNPP SSCs. Pertinent to this evaluation, this activity will include installation of a power-operated gate across the ISFSI haul road adjacent to the NSF Sallyport. This activity results in changing the ISFSI USAR as follows (with deletions lined through and additions underlined):*

1) Change USAR Volume I, Section 10.3.4.1, Item B. Specifications, first paragraph (as revised by 72.48 #94-0-101-003, which is scheduled to be included in the 1995 USAR revision) to read:

"The roadway or ground surface elevation perpendicular to the route to or from the ISFSI within an 8.0 ft proximity of the transfer trailer shall not be more than 20 inches below the trailer road surface centerline elevation. The paved portion of the road shall be a minimum of 16 feet wide and the adjacent paved, gravel or soil shoulder shall extend to make the transfer route at least 28 feet wide. The lowest point within the 28 foot wide transfer route shall not be lower than 20 inches below the road centerline and may contain typical roadside features, including curbs, fences, guard rails and light poles which do not constitute potential puncture mechanisms for the cask. The shoulders may not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture mechanism. The components associated with the vehicle barrier system, installed adjacent to the Nuclear Security Facility and closing the 16 foot wide ISFSI haul road at the Protected Area boundary, have been analyzed and do not represent a puncture risk to the transfer cask. The road shall be closed to other vehicles when transporting spent fuel."

Reason for the Activity:
The current ISFSI USAR describes the transfer route and restricts items which could present a risk of transfer cask (TC) and Dry Shielded Canister (DSC) puncture from placement within the 28 foot wide transfer route. Without clarification, this restriction could be interpreted to include vehicle barrier components, such as barrier support buttresses, and could lead to unnecessary concern or confusion about site compliance with the ISFSI USAR. The installation of vehicle barriers across the ISFSI haul road is necessary to meet the requirements of 10CFR73.55. The proposed vehicle barrier buttresses have been shown by calculation 95-0185 to be enveloped by the existing cask drop analysis. In addition, the consequences of an uncontrolled drop of the vehicle barrier's crash beam has been shown by the same calculation to be enveloped by the existing cask drop analysis.

Function(s) of Affected SSCs:
The ISFSI haul road provides a hard paved surface for the tractor to transport spent fuel in a NUHOMS-24P DSC/TC from the Auxiliary Building to the ISFSI.

SAR Sections Reviewed:
ISFSI Vol. I, All Sections;
ISFSI Vol. IV, Section 2, SAR Q&A December 20, 1990;
ISFSI Vol. IV, Section 4, NRC ISFSI SER November 1992;
ISFSI Vol. V, All Sections.

ATTACHMENT 3, SAFETY EVALUATION FORM

(Page 3 of 4)

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

Yes [✓ No] May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

The equipment important to safety is the spent fuel haul rig (TC/DSC mounted on the transfer trailer/support cradle and pulled by the tractor). The malfunction of this equipment involves the sequence of events which could lead to a cask drop. The scenario is comprised of: (1) the haul rig veers off course; (2) the transfer trailer strikes a roadside object and is damaged; (3) the damage causes the transfer trailer to tip far enough to drop the TC/DSC; and, (4) the TC/DSC hits something. The malfunction of concern is the loss of directional control of the transfer rig. Items 2, 3, and 4 are subsequent steps with a cause-and-effect relationship leading to the consequence of concern, TC puncture, which is addressed in the consequences section, below. The transport vehicle is administratively controlled to stay in the center of the transfer route and at very low speed. In addition, the paved road is at least 16' wide and provides several feet of margin in the event of a loss of vehicle control. The vehicle barrier buttresses are 24' apart and do not encroach upon the 16' transfer road (do not reduce the margin for correcting vehicle misdirection). The probability of loss of vehicle control is independent of the presence of the proposed vehicle barrier across the haul road. The administrative controls in place are sufficient to ensure the vehicle does not veer off course. Hence, the probability of occurrence of a malfunction of equipment important to safety previously evaluated is not increased.

Yes [✓ No] May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

The consequences of an accident or malfunction in the TC/DSC are associated with a cask drop leading to puncture of the TC/DSC and release of the enclosed fission products to the atmosphere. Calculation C-95-0185 demonstrates that a cask drop onto the vehicle barrier buttresses does not lead to a cask puncture. Hence, the consequences of a malfunction are not increased.

Yes [✓ No] May the probability of occurrence of an accident previously evaluated in the SAR be increased?

The applicable accident previously evaluated is the drop of the TC/DSC for heights up to 80 inches above a thick hard surface. The probability of a cask drop accident is not increased because the physical dimensions and operation of the spent fuel haul rig (TC/DSC mounted on the transfer trailer/support cradle and pulled by the tractor) do not change.

Yes [✓ No] May the consequences of an accident previously evaluated in the SAR be increased?

The consequences of a TC/DSC drop deal with dose from release of fission products via a puncture of the TC/DSC. BGE Calculation 95-0185 provides the parameters between which the TC/DSC integrity during a cask drop accident onto the vehicle barrier is assured. The required buttress dimensions have been incorporated into the modification Design Instructions. Fuel moves will be restricted if the above-ground portions of the barrier buttresses are in an intermediate stage of completion. This restriction is stated in the Design Instructions. Excavation restrictions have also been incorporated into the modification Design Instructions to ensure the 80 inch height restriction is not exceeded should fuel moves occur during the mod implementation period. Since the physical dimensions and operation of the TC/DSC and trailer/support system do not change due to the presence of the proposed vehicle barrier and because of the prescribed dimensions of the barrier buttresses, puncture of the TC/DSC will not occur and the consequences of a cask drop are not increased.

ATTACHMENT 3, SAFETY EVALUATION FORM

(Page 4 of 4)

2. The possibility for an accident or malfunction of a different type than evaluated previously in the SAR is not created.

__ Yes _✓_ No May the possibility of a malfunction of a different type than previously evaluated in the SAR be created?

Any malfunction of the TC/DSC would be associated with a drop height greater than 80 inches. Since the physical dimensions and operation of the TC/DSC and trailer/support system prevent a fall of over 80 inches, which is currently acceptable and does not change, the possibility of a new malfunction is not increased.

__ Yes _✓_ No May the possibility of an accident of a different type than previously evaluated in the SAR be created?

The proposed changes affect transportation of spent fuel inside the TC/DSC. The configuration of the proposed gate is a semaphore-style gate with a reinforced steel crash beam and counterweight. The effects of the gate dropping on the TC have been shown to be within the existing cask drop analysis (BGE Calc 95-0185). Since the bounding case envelopes the proposed activities, no possibility of a new accident is created.

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

__ Yes _✓_ No Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Tech Spec Basis 2.3 states that the TC drops less than 80 inches will not produce unacceptable damage to the TC/DSC. Analysis of the proposed barrier buttresses (for a cask drop) and crash beam (for a barrier crash beam drop onto the TC) show that the effects on the TC and DSC are within the envelope of the current design bases (BGE Calc 95-0185).

Complete for a 72.48:

__ Yes _✓_ No Will the proposed activity involve a significant increase in occupational dose?

The opening time for the proposed gate is less than 30 seconds and may be performed in a manner which will not delay spent fuel transport operations. Therefore, there will be no significant increase in occupational dose associated with the addition of this vehicle barrier.

__ Yes _✓_ No Will the proposed activity involve a significant unreviewed environmental impact?

Since the transfer route does not change, adding the proposed vehicle barrier does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

The Independent Spent Fuel Storage Installation (ISFSI) haul road provides a hard paved surface for the tractor to transport spent fuel in a NUHOMS-24P DSC/TC from the CCNPP Auxiliary Building to the ISFSI. The ISFSI USAR description of the transfer route was changed to allow the presence of a vehicle barrier to be installed to comply with 10CFR73.55, as amended in August, 1994. The change allows the vehicle barrier's supporting buttresses to be installed within the 28 foot wide transfer route. It has been confirmed by calculation that a cask drop onto the vehicle barrier buttresses and a crash beam drop onto the TC are enveloped by the existing cask drop analysis. This change does not constitute an unreviewed safety question, a change to the Technical Specifications or Bases, a significant increase in occupational exposure nor an unreviewed environmental impact for the ISFSI.

ATTACHMENT 2, UFSAR CHANGE REQUEST FORM (UCR) (Page 1 of 2)

NONMOD # 95-028

UCR # ______ For LU use only

To: UFSAR Coordinator
From: MATTHEW A. CARR (Printed Name) Work Group CEU Date 6/1/95
Phone Number: 260-6848 System Number 107

SECTION 1 (Change Initiation)

A. UFSAR CHANGE SOURCE DOCUMENT

[✓ FCR]/FEC/MCR# 95-0201 (Circle One) Procedure # ES19950 1089-000

License Amendment # ______

Regulatory Generic Correspondence # ______
Generic Letter, Bulletin or Information Notice

Unit 1 ___ Unit 2 ___ Common ___ ISFSI X

B. SAFETY EVALUATION [Check One]

___Safety Evaluation Screening Not Required per Attachment 5 Criteria: ______
BASIS FOR TYPE 1 UFSAR CHANGE CLASSIFICATION
(Attach additional pages, if required)

Is the proposed UFSAR change consistent with the Technical Specifications/License Conditions or Bases? _____ Yes _____No
(If the above question is answered "No", consult CCI-143 for License Amendment Proposals.)

___Safety Evaluation Screening attached SE00001
X Safety Evaluation Log# (attached) 95-0001 (72.48)
___NRC Safety Evaluation Report (SER) attached, dated ______

C. DESCRIPTION OF UFSAR CHANGE:

INSERT A STATEMENT SPECIFICALLY RECOGNIZING THAT THE VEHICLE BARRIER TO BE INSTALLED ACROSS THE ISFSI HAUL ROAD DOES NOT REPRESENT A PUNCTURE THREAT TO THE TRANSFER CASK IN THE UNLIKELY EVENT OF A CASK DROP ACCIDENT DURING FUEL TRANSFER OPERATIONS

D. UFSAR SECTIONS AFFECTED: [Attach Marked up Page(s)]

VOL I, SECTION 10.3.4.1, ITEM B. SPECIFICATIONS. (AND Q+A SECTION)

ATTACHMENT 2, UFSAR CHANGE REQUEST FORM (UCR) (Page 2 of 2)

Source Document #: 95-028

UCR#: ______ For LU use only

SECTION 2 (Interdisciplinary Reviews)

VERIFICATION THAT THE TECHNICAL CONTENT OF THIS UFSAR CHANGE AGREES WITH THE FACILITY DESIGN AND CONFIGURATION

RESP. IND. G. Tesfaye Getachew WORK GROUP: Licensing 7/5/95
Printed Name and Signature

RESP. IND. S. Mixett WORK GROUP: PSEU 7/6/95
Printed Name and Signature

RESP. IND. R.H. Beall / R.H. Beall WORK GROUP: NFM 7/10/95
Printed Name and Signature

SECTION 3 (Implementation Verification Prior to UFSAR Incorporation)

VERIFICATION THAT NOTIFICATION HAS BEEN RECEIVED INDICATING THAT PLANT MODIFICATION INCORPORATED:

☐ Partial Implementation ☐ Unit 1 ☐ Unit 2

This change will be incorporated in Revision No. ______

UFSAR COORDINATOR: ______ DATE: ______

SECTION 4 (Final Review/Approval Prior to UFSAR Incorporation)

MODIFICATIONS - VERIFICATION THAT THE UFSAR CHANGE IS IN AGREEMENT WITH CURRENT DESIGN INFORMATION

NONMODS - VERIFICATION OF CONCURRENCE WITH THE BASIS FOR CLASSIFYING THE CHANGE AS A TYPE 1 UFSAR CHANGE AND THE DETERMINATION OF CONSISTENCY WITH THE TECHNICAL SPECIFICATIONS (IF APPLICABLE), AND THAT THE TECHNICAL CONTENT OF THIS UFSAR CHANGE AGREES WITH THE FACILITY DESIGN AND CONFIGURATION.

X

RESPONSIBLE ENGINEER: M. A. CARR DATE: 8/10/95
K.C. Bostic 8/10/95

RESP. ENGR'S. SUPERVISOR: MJ Gahan DATE: 8.17.95

SECTION 5 (Implementation Review)

VERIFICATION THAT THE DOCUMENTATION REQUIRED BY CCI-177 IS INCLUDED IN THE UFSAR LICENSING PACKAGE AND THAT THE UFSAR CHANGE HAS BEEN ACCURATELY INCORPORATED.

UFSAR COORDINATOR: ______ DATE: ______

10.3.4 LIMITING AND OPERATING CONDITIONS FOR TRANSFER CASK CONTAINING LOADED DSC

10.3.4.1 Transfer Route Selection [See Reference 10.2]

A. Title: Transfer Route Selection

B. Specifications: The roadway or ground surface elevation perpendicular to the route to or from the ISFSI within an 8.0 ft proximity of the transfer trailer shall not be less than that of the trailer road surface elevation as measured at the outer edge of asphalt pavement. The paved portion of the road shall be a minimum of 16 feet wide and the adjacent paved, gravel or soil shoulder shall be a minimum of 7 feet wide on each side of the road. The shoulder shall be level with or higher than the outer edge of the pavement and may contain typical roadside fixtures, including curbs, fences, guard rails and light poles which do not constitute potential puncture devices for the cask. The shoulders may not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture device. The road shall be closed to other vehicles when transporting the spent fuel.

SEE NOTE

The maximum drop height of the cask from the transfer trailer to the roadbed does not exceed 80 inches.

C. Applicability: This specification is applicable to DSC transfer utilizing the NUHOMS-24P transfer cask and trailer.

D. Objective: Ensure that a potential drop height of 80 inches is not exceeded.

E. Action: Repair the road to its proper elevation.

F. Surveillance: Prior to the transfer of a DSC to or from an HSM, the proposed transfer route shall be visually inspected.

G. Bases: A drop from a height of 80 inches or less does not compromise the design margins of the transfer cask or DSC.

NOTE - THE SUBJECT PARAGRAPH WAS REVISED, BUT NOT YET INCORPORATED INTO THE USAR. THE REVISING 72.48 PAGES ARE ATTACHED (72.48 # 94-0-101-003).

ATTACHMENT 3, SAFETY EVALUATION FORM

Page 1 of 4

ACTIVITY: Calvert Cliffs ISFSI USAR Change 50.59 Log No.____ or 72.48 Log No. 94-0-101-003

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

___YES _X_ NO Involve an Unreviewed Safety Question (USQ)?
___YES _X_ NO Involve a change to the Technical Specifications/License Conditions or Bases?
_X_YES ___ NO Require a change or addition to the UFSAR or USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

___YES _X_ NO Involve a Significant Increase in Occupational Dose?
___YES _X_ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: Sam Shakir SAM SHAKIR Department: CCSO Date: 8/2/94
PRINTED NAME AND SIGNATURE (VECTRA)

_X_YES ___NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: J.B. MAKAR PRINTED NAME	Resp. Ind.: G. Tesfaye PRINTED NAME	Resp. Ind.: PRINTED NAME
K4042 SIGNATURE	SIGNATURE	SIGNATURE
Work Group: System Engineer	Work Group: Licensing Unit	Work Group:
Date: 8-5-94	Date: 9-20-94	Date:

Approved ✓ Disapproved ___	Approved ✓ Disapproved ___
Signature Sam Shakir for Mases Taylor INDEPENDENT REVIEWER (VECTRA) Per Tele. Con.	Signature Michael J. Gahan GS-DES, GS-TSES, or PE-PDSU M. J. GAHAN CCSO 8/10/94 9.20.94
Date: 8/4/94	Date:

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 94-145 Date: 9-28-94

Recommend Approval ✓ Recommend Disapproval___ Signature_____ Date: 9-28-94
POSRC CHAIRMAN

Approved ✓ Disapproved___ Signature_____ Date: 9/28/94
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.
OSSRC Meeting No.:__________ Date:__________

Recommend Approval____ Recommend Disapproval____ Signature__________ Date:__________
OSSRC CHAIRMAN

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: Calvert Cliffs ISFSI USAR Change 50.59 Log No.____ or 72.48 Log No. 94-0-101-003

Proposed Activity:
This activity changes the requirements for the ISFSI transfer route to allow the shoulders to be up to 20" lower than the centerline elevation of the road surface. This activity results in changing the ISFSI USAR as follows:

1) Change USAR Volume IV, Section 2 USAR Q&A, Question 8.0-5 Response, first paragraph to read:

"The transfer cask will be transported along an asphalt or concrete paved road which is at least 16 feet wide and which has shoulders which extend to make the transfer route at least 28 feet wide. The road is approximately 3,300 linear feet with grades which range from 0% to 3% except for an approximate 50 foot length which carries a 5.7% grade. The roadbed is level except for a negligible 1% slope required to create a crown in the road for drainage and a transverse slope at any point along the transportation route of less than 10%. The shoulders are either level with the road, or slope down from the road such that the maximum vertical distance from the centerline of the road to the lowest point within the 28 foot wide transfer route is 20 inches. In those locations where the paved road abuts up to existing blacktop, or concrete paving, the shoulder is discontinued. The shoulder may be paved, gravel or soil and contain typical roadside fixtures, including curbs, fences, guard rails and light poles which do not constitute potential puncture mechanisms for the cask during a drop. The shoulders do not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture mechanism during a cask drop. For the entire route that the transfer cask is transported there will exist a minimum 8 foot wide zone on each side of the trailer that is not more than 20 inches below the road centerline elevation."

2) Change USAR Volume I, Section 10.3.4.1, Item B. Specifications, first paragraph to read:

"The roadway or ground surface elevation perpendicular to the route to or from the ISFSI within an 8.0 ft proximity of the transfer trailer shall not be more than 20 inches below the trailer road surface centerline elevation. The paved portion of the road shall be a minimum of 16 feet wide and the adjacent paved, gravel or soil shoulder shall extend to make the transfer route at least 28 feet wide. The lowest point within the 28 foot wide transfer route shall not be lower than 20 inches below the road centerline and may contain typical roadside fixtures, including curbs, fences, guard rails and light poles which do not constitute potential puncture mechanisms for the cask. The shoulders may not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture mechanism. The road shall be closed to other vehicles when transporting the spent fuel."

Reason for Activity:
The current ISFSI USAR description of the transfer route and shoulders is unnecessarily restrictive regarding the allowable elevation of the shoulder surface relative to the transfer road surface and the relative width of the paved road and the adjacent shoulders. The current description of the road specifies the elevation of the shoulder surface to be not less than that of the trailer road surface centerline elevation. This description is restrictive considering that the shoulders are affected by heavy rain and at times get eroded and washed away requiring constant repair. The significance of the shoulder elevation is to limit the drop height of the cask to its designed limit of 80 inches. Since the maximum distance from the bottom of the transfer cask to the road centerline is 56.25 inches, this allows the lowest point on the transfer route to be up to 20 inches below the elevation of the road centerline without affecting the design basis of 80 inches. The current description of the shoulders width is also restrictive. The ISFSI USAR describes the shoulders as being a minimum of 7 feet wide on each side of the road. This will now be changed to specify a total width of the transfer route including shoulders at a minimum of 28 feet.

Function (s) of affected SSC:
Transport road provides a hard paved surface for the tractor to transport spent fuel in a NUHOMS®-24P canister/transfer cask from the Auxiliary Building to the ISFSI.

ISFSI USAR Sections Reviewed:
Vol.. IV, Section 2; Vol. I, Section 4.1.1; Vol. I, Section 10.3

1) Change USAR Volume I, Section 10.3.4.1, Item B. Specifications, first paragraph (as revised by 72.48 #94-0-101-003) to read:

"The roadway or ground surface elevation perpendicular to the route to or from the ISFSI within an 8.0 ft proximity of the transfer trailer shall not be more than 20 inches below the trailer road surface centerline elevation. The paved portion of the road shall be a minimum of 16 feet wide and the adjacent paved, gravel or soil shoulder shall extend to make the transfer route at least 28 feet wide. The lowest point within the 28 foot wide transfer route shall not be lower than 20 inches below the road centerline and may contain typical roadside features, including curbs, fences, guard rails and light poles which do not constitute potential puncture mechanisms for the cask. The shoulders may not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture mechanism. The components associated with the vehicle barrier system, installed adjacent to the Nuclear Security Facility and closing the 16 foot wide ISFSI haul road at the Protected Area boundary, have been analyzed and do not represent a puncture risk to the transfer cask. The road shall be closed to other vehicles when transporting spent fuel."

ATTACHMENT 3, POSRC/PRC PRESENTATION FORM

POSRC/PRC PRESENTATION FORM

Presentation Date:

Presenter: **M. A. Carr** Extension: **6848**

Procedure or Activity: **ISFSI USAR Change due to Vehicle Barrier System Upgrade Modification (ES199501089-000)**

Purpose of Presentation:
☑ Recommendation for Approval
☑ Information
☐ Close OI
☐ Extend OI

Activity Summary: (See POSRC/PRC Presenter's Guide III.A.1): **ES199501089 (FCR 95-0201) will upgrade the site's vehicle barrier systems to prevent access by a bomb carrying malevolent vehicle within the safe standoff distance of selected SSCs. All of the new barriers will be outside the Protected Area and are NSR. One barrier will cross the ISFSI haul road, adjacent to the NSF Sallyport. The ISFSI haul road has restrictions on the types of items which may be installed within the 28 foot wide transfer route. No items which represent a puncture mechanism to the spent fuel transfer cask (TC) may be installed in this zone. BGE Calculation C-95-0185 shows the required dimensions for the barrier not to pose a puncture risk to the TC. The proposed ISFSI USAR change explicitly provides for the proposed vehicle barrier to be present within the ISFSI haul road's 28 foot width.**

In addition, spent fuel moves are scheduled throughout the construction period. Restrictions are proposed for the construction period to ensure the limitations of the ISFSI USAR are not violated during fuel movements durng the construction period.

Safety Issues Involved: (See POSRC/PRC Presenter's Guide II.B, C, D, E, and III.A.2): **The design basis accident is the drop of the TC while moving spent nuclear fuel. BGE Calculation C-95-0185 shows that the loads imposed by the proposed barrier components are enveloped by the existing TC drop analysis.**

Recommendations to POSRC or PRC: (See POSRC/PRC Presenter's Guide II.F, G, H, and III.A.3 and F):
Recommend approval of the Safety Evaluation and USAR change, with the following precautionary actions:
(to NFM) Spent fuel transfers to or from the ISFSI will be prohibited from the time the construction of the ISFSI haul road vehicle barrier adds any above-ground component until construction of the components within the 28 foot wide transfer route is essentially complete.
(to Project Management) Work progress will be coordinated by the Project with NFM to minimize disruption of scheduled fuel moves.
(to Project Management) Excavations within the 28 foot transfer route may not exceed 20 inches during the period of any fuel moves; and, no tools or equipment which could represent a puncture threat to the TC may be present within the 28 foot transfer route during spent fuel moves.

SAFETY EVALUATION FORM
(Attachment 3)

Page 1 of 4

ACTIVITY: ES199600014	**50.59 LOG NO: XXXXX**	**72.48 LOG NO: SE00002**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ Yes ☒ No Involve an unreviewed safety question (USQ)?

☐ Yes ☒ No Involve a change in the Technical Specifications/License Conditions or Bases?

☒ Yes ☐ No Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations:

☐ Yes ☒ No Involve a significant increase in occupational dose?

☐ Yes ☒ No Involve a significant unreviewed environmental impact?

Prepared by: M. A. Carr G.V PATEL GV Patel *(Printed Name and Signature)*	Department: NED/DES/CEU	Date: ~~2/16/96~~ 2/20/96

☒ Yes ☐ No Is a special review required by groups other than the group to which the Preparer belongs?

(Printed Name and Signature)

Responsible Indiv:	C. G. Sarau Charles Sarau	Work Group: Facilities Svcs	Date: 2/20/96
Responsible Indiv:	E. M. Tyler Em Tyler	Work Group: Licensing	Date: Feb 16, 96
Responsible Indiv:		Work Group:	Date:
Independ reviewer:	K.C. ANSTEE KC Anstee	☒ Approved ☐ disapproved	Date: 2/21/96
GS-DES/GS-TSES/PE-PDSU	MICHAEL J. GAHAN Michael J. Gahan	☒ Approved ☐ disapproved	Date: 2/22/96
The POSRC has reviewed this evaluation to NS-2-101. **Recommend:** ☑ Approval ☐ Disapproval		POSRC Meeting No.: 96-17 John T Carroll **POSRC Chairman**	Date: 2-26-96 Date: 2-26-96
Recommend: ☑ Approval ☐ Disapproval		P.E. Katz **Plant General Manager**	Date: 2/26/96
The OSSRC has reviewed this evaluation to NS-2-100. **Recommend:** ☒ Approval ☐ Disapproval		OSSRC Meeting No.: SES J S Russell **OSSRC Chairman**	Date: 7-1-96 Date: 7-2-96

EN-1-102
Rev 2
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FORM MEETS INTENT OF REV 3 maCarr 2/16/96

ACTIVITY: ES199600014	**50.59 LOG NO:** XXXXX	**72.48 LOG NO:** SE00002

Proposed Activity: The underground storage tanks at the heavy duty lube shop were replaced by new underground storage tanks (USTs) at the Transportation Facility (TF) when the lube shop was demolished to facilitate construction of the Nuclear Office Facility (NOF). These new tanks are two 4000 gallon tanks for gasoline and diesel fuel and one 550 gallon tank for storage of waste oil.

Reason for Activity: This 72.48 evaluates the location of the USTs, which is closer to the ISFSI haul road and larger than stated in correspondence to the NRC (now part of the USAR in Appendix A, Q&A). The original USTs were approximately 200 feet from the spent fuel transfer route. The current location is approximately 70 feet from the transfer route. The USTs were described as two 3000 gallon tanks. The new USTs are two 4000 gallon tanks and one 550 gallon tank.

Function(s) of affected SSC: The affected SSC is the ISFSI spent fuel transfer route. This route is used to transport spent nuclear fuel in the Transfer Cask and Dry Shielded Canister from the CCNPP Aux Building to the ISFSI.

SAR Sections Reviewed: ISFSI USAR Vols I, III, and IV.

Complete 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

☐ Yes ☒ No May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction: The pre-license Q&A correspondence and the Safety Evaluation Report acknowledged the presence of the original refueling depot. However, the evaluation found underground storage of fossil fuels meeting NFPA 30-1987, *Flammable and Combustible Liquid Code,* was not of concern, but a tanker truck carrying fossil fuels represented a risk to be avoided. The consequences of a fossil fuel carrying tanker truck induced fire or explosion accident have not been analyzed for the transfer cask. As a result, restrictions were placed on the allowed location (>100 meters from transfer route) and movement of tanker trucks inside the plant main entrance (no movement allowed) while spent fuel transfer operations are in progress. These restrictions are not changed due to the relocation of the TF. None of the accidents or malfunctions of equipment important to safety evaluated in the SAR involve the TF USTs. Therefore, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes ☒ No May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction: See the answer, above.

☐ Yes ☒ No May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident: See the answer, above.

☐ Yes ☒ No May the consequences of an accident previously evaluated in the SAR be increased:

Consequences of Accident: See the answer, above.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

☐ Yes ☒ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

ACTIVITY: ES199600014	**50.59 LOG NO: XXXXX**	**72.48 LOG NO: SE00002**

Probability of New Malfunction: The USAR analyzed the code-required stand-off distance for USTs (NFPA 30-1987, *Flammable and Combustible Liquid Code*). Underground storage of flammable and combustible liquids is considered the safest form of storage, The NFPA-specified minimum distance is 25 feet. The refueling depot dispensing pumps, USTs and their tank vents are all approximately 70 feet, or further, from the nearest side of the ISFSI spent fuel transfer route.

☐ Yes ☒ No May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of a New Accident: See the answer, above.

COMPLETE FOR 50.59 AND 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ Yes ☒ No May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

BASES	DISCUSSION OF WHY THE MARGIN OF SAFETY IS NOT REDUCED
3/4.5 Fire Protection	The basis acknowledges the proximity of the refueling depot and reiterates the objective of the Tech Spec is to preclude an accident involving fire or explosion near the TC due to a large amount of fossil fuels. The preclusion of tanker trucks within 100 meters ensures there will be no tanker truck at the TF during spent fuel moves to the ISFSI.

COMPLETE FOR 72.48:

☐ Yes ☒ No Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose: Relocating the TF did not change the spent fuel transfer route, therefore, there are no delays in spent fuel transfer operations which would increase occupational dose due to the location of the TF.

☐ Yes ☒ No Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact: Changing the location of the TF and increasing the UST sizes by such a small amount (1000 gallons each) does not represent a significant unreviewed environmental impact. In addition, the TF was permitted by Calvert County under their building and environmental permitting process. Any environmental impacts caused by TF construction were addressed under that permitting process.

SUMMARY: (For NRC Report, provide a brief overview)

The location of the Transportation Facility was changed during construction of the Nuclear Office Facility (NOF) to a location east of the ISFSI spent fuel transfer route. The new location is closer to the transfer route than stated in the SAR (Appendix A, Q&A, Question 8.0-6), but still outside the NFPA 30-1987 specified setback of 25 feet. As well, the size and number of underground storage tanks was increased from two 3000 gallon tanks to two 4000 gallon tanks and one 550 gallon tank for diesel fuel, gasoline, and waste oil, respectively. This change does not represent a USQ because the USTs are still outside the NFPA setback requirements. In addition, the new location is such that the 100m tanker truck exclusion zone will preclude fuel deliveries during the time of spent fuel transfer operations from the CCNPP Aux Building to the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

Independent Spent Fuel Storage Installation Safety Evaluation **72.48 Log No: SE00003**
ES199601368 **Supplement 000** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
YES	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk PRINTED NAME AND SIGNATURE Department: NED-CEU 42-01-04 Date: 8-30-96

DE Reviewer: J. N. Woodfield Jon N. Woodfield Department: NED-CEU 42-01-04 Date: 8/30/96

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. L. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye GT 8/30/96 SIGNATURE / DATE	8/30/96 SIGNATURE / DATE	R H Beall 8/30/96 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: M.A. CARR
INDEPENDENT REVIEWER

Date 8/30/96

[✓ Approved] Disapproved

Signature: Michael J Graham
for [✓ GS-DES] GS-TES, or PE-PDSU

Date 8.30.96

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 96-110[redacted] JEG Date: 8.30.96

Recommend Approval ✓ Recommend Disapproval ______ Signature: John Carll Date 8-30-96
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: John Carll Date 8-30-96
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

OSSRC Meeting No.: .: 97-01 Date: 1/16/97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date: ______
OSSRC CHAIRMAN

ATTACHMENT 3, SAFETY EVALUATION FORM

Independent Spent Fuel Storage Installation Safety Evaluation		**72.48 Log No: SE00003**	
ES199601368	**Supplement 000**	**Revision 0000**	**Page 2 of 5**

Proposed Activity: A technical review of ISFSI documentation that was submitted to and received by the NRC in 1992, but was never reviewed by the NRC, detected a discrepancy that will require a revision to the ISFSI USAR

Proposed ISFSI USAR Change: Change the description of the DSC insertion as described in Section 4.2.3.2 to reflect the deletion of dry lubricant from the DSC shell and the addition of Nitronic hard sliding rails to the TC and HSM. This change was fully evaluated and justified in 1991 by Pacific Nuclear Services, Inc., and approved by BGE for construction.

Reason for ISFSI USAR Change: The DSC is designed to slide from the TC into the HSM and back without undue galling, scratching, gouging, or other damage to the sliding surfaces. Substantial galling had been observed in a similar application of the dry lubricant to the DSC shell. The addition of the Nitronic rails was made as a design improvement, and testing in similar applications was found to perform substantially better than the previous design. BGE approved this design change for construction in 1991. The ISFSI license was issued in November of 1992, and ISFSI loading operations began in November of 1993. All ten fuel moves to date have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are three major components of the NUHOMS-24P system that are addressed in this safety evaluation. Those three components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); and 3) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those three components.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules will be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DCS contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for temporary storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide temporary storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

SAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.4, 3.6, 4.2, 4.7, 5.1, 7.4, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

Independent Spent Fuel Storage Installation Safety Evaluation 72.48 Log No: SE00003
ES199601368 Supplement 000 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. The possible malfunction for the DSC insertion would involve the complete stoppage of the insertion process due to undue galling, scratching, gouging, and damage to other sliding surfaces. The proposed USAR change involves the deletion of dry lubricant from the DSC shell and the addition of Nitronic hard sliding rails to the TC and HSM. As such, the rails are coated with dry film lubricants in lieu of the DSC. Similar applications at other ISFSI sites have been seen to perform substantially better than the previous design. In addition, since ISFSI loading operations began in November of 1993, all ten fuel moves to date have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces. This is considered a design improvement which will reduce the probability of a DSC insertion malfunction.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. The consequences of a complete stoppage of the DSC insertion would result in placing the DSC safely back into the TC. The proposed USAR change is a design improvement which would allow the restoration process to occur in a more timely manner. As such, the consequences of a DSC insertion malfunction would not be increased.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of this proposed activity. One accident scenario described in the ISFSI USAR assumes that the spent fuel rods and the DSC pressure boundary are ruptured and leakage occurs due to an event of unspecified origin. The origin of rupture during the DSC insertion process would be the sliding surfaces. It has been previously stated that the proposed USAR change involves the deletion of dry lubricant from the DSC shell and the addition of Nitronic hard sliding rails to the TC and HSM. This change, which occurred in 1991, was found to perform better than the previous design at other sites. In addition, this design has resulted in ten successful spent fuel moves. Most notably, the Nitronic hard sliding rails have provided a mechanism for the smooth, damage free transfer of our DSC's from the TC to the HSM. Since the probability of damage to the DSC via the DSC transfer process has been reduced, the probability of occurrence of the DSC leakage accident previously evaluated in the ISFSI USAR will not be increased.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. There are no structural or thermal consequences, and only minimal radiological consequences resulting from the DSC leakage accident as described in the ISFSI USAR. Since the design change has resulted in a smooth, damage free operation, no potential consequences are introduced that could increase the consequences of the DSC leakage accident described in the ISFSI USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

Independent Spent Fuel Storage Installation Safety Evaluation 72.48 Log No: SE00003
ES199601368 Supplement 000 Revision 0000 Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. The addition of the Nitronic hard sliding rails, which are ¼" thick and 3" wide, to the existing support rails, has been evaluated by structural calculations to have no adverse impact on the structural adequacy of the ISFSI design.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. No new accident scenarios are created as a result of the addition of the Nitronic hard sliding rails to the TC and HSM.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The design change was an improvement to the transfer operation of the DSC from the TC to the HSM, and as such, does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

Independent Spent Fuel Storage Installation Safety Evaluation 72.48 Log No: SE00003
ES199601368 Supplement 000 Revision 0000 Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: A technical review of ISFSI documentation that was submitted to and received by the NRC in 1992, but was never reviewed by the NRC, detected a discrepancy that will require a revision to the ISFSI USAR

Proposed ISFSI USAR Change: Change the description of the DSC insertion as described in Section 4.2.3.2 to reflect the deletion of dry lubricant from the DSC shell and the addition of Nitronic hard sliding rails to the TC and HSM. This change was fully evaluated and justified in 1991 by Pacific Nuclear Services, Inc., and approved by BGE for construction.

Reason for ISFSI USAR Change: The DSC is designed to slide from the TC into the HSM and back without undue galling, scratching, gouging, or other damage to the sliding surfaces. Substantial galling had been observed in a similar application of the dry lubricant to the DSC shell. The addition of the Nitronic rails was made as a design improvement, and testing in similar applications was found to perform substantially better than the previous design. BGE approved this design change for construction in 1991. The ISFSI license was issued in November of 1992, and ISFSI loading operations began in November of 1993. All ten fuel moves to date have resulted in a smooth transfer of the DSC from the TC into the HSM without any damage to the sliding surfaces.

Activity Summary: After a thorough and intense review, it has been concluded that the proposed activity:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199601328-001 **50.59 Log No.:** **72.48 Log No.: SE00004**

Based on the attached discussion, does this activity:

<u>Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations</u>

√ YES NO Involve an unreviewed safety question (USQ)?
YES √ NO Involve a change in the Technical Specifications/License Conditions?
√ YES NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

<u>Applicable to 10 CFR 72.48 Safety Evaluations</u>

YES √ NO Involve a Significant Increase in Occupational Dose?
YES √ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: K. C. Anstee K.C. Anstee Department: DES-CEU Date: 01/25/99
PRINTED NAME AND SIGNATURE

√ YES NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: J. L. Stone Work Group: Reliability Eng.	Resp. Indv.: A. L. Simpson Work Group: Licensing	Resp. Indv.: R. H. Beall Work Group: Nuclear Fuels
JLStone 1/25/99 **SIGNATURE / DATE**	1-26-99 **SIGNATURE / DATE**	R H Beall 1-26-99 **SIGNATURE / DATE**

[✓ Approved] Disapproved
Signature J.E. Remeniuk JER
INDEPENDENT REVIEWER
Date 1.27.99

[✓ Approved] Disapproved
Signature Michael J. Gekara
for ~~GS-DES,~~ GS-TES, or PE-PDSU
Date 1-27-99

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 98-008 Date: 2-3-98

Recommend Approval ✓ Recommend Disapproval ______ Signature ______ Date 2-3-98
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature ______ Date 2/4/98
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes X No ______
Signature: ______ Date: 5/13/99
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: 99-03

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199601328-001 **50.59 Log No.:** **72.48 Log No.: SE00004**

Proposed Activity: Section 2.2.1.1 of the Independent Spent Fuel Storage Installation (ISFSI) USAR is related to information on aircraft and their flight paths for Patuxent River Naval Air Station. The above noted section is outdated and will be updated under this activity.

Reason for Activity: The purpose of this activity is to revise Section 2.2.1.1 of the ISFSI USAR to reflect the current information on aircraft and their flight paths for Patuxent River Naval Air Station.

Function(s) of affected SSC: This change affects the entire Independent Spent Fuel Storage Installation.

ISFSI USAR Revision No.: 7 Tech Spec Bases Rev. No.: 1

ISFSI USAR Sections reviewed: Chapter 2, 3, 8, and the electronic docket. Tech Spec Bases Reviewed: Entire Bases for Sections 2.0 and 3/4.0

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

YES √ NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

Aircraft hazard is an external event which is not specifically addressed or identified within the Chapter 8 accident analysis. Section 2.2 of the ISFSI USAR provides a description of existing airports, a description of some of the aircraft using them, weight of the heaviest aircraft at Patuxent River Naval Air Station, the number of take-offs and landings, and flight paths. Within this description of airports it is noted that aircraft at Patuxent River Naval Air Station would come no closer than seven miles to the ISFSI.

The actual aircraft hazard during original construction and licensing of the ISFSI was never quantified. This was due to the fact that the aircraft conditions were the same for both the ISFSI and CCNPP along with the fact that aircraft hazard for CCNPP (which was also never quantified) was judged to be acceptably low by the NRC at the time of construction and licensing of CCNPP. Section 3.1.2 of the Safety Evaluation by the Directorate of Licensing U.S. Atomic Energy Commission in the Matter of BGE CCNPP Units 1 & 2 dated 8/28/72 stated the following:

> "Considering the relatively small number of aircraft movements at these airports and their distances from the Calvert Cliffs site, the applicant concluded and we concur, that the probability of an aircraft crash affecting the plant is so low that no special design provisions should be made in the plant for such an event."

The above statement implies that the probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines was less than 10^{-7} per year. Regulatory Guide 1.70 (Reference 1), which is utilized herein as a guideline (BGE is not committed to the Reference 1 Regulatory Guide), states that if the probability of an accident is on the order of 10^{-7} per year or greater, the accident should be considered a design basis event, and a detailed analysis of the effects of the accident on the plant's safety-related structures and components should be provided.

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199601328-001 **50.59 Log No.:** **72.48 Log No.: SE00004**

Probability of Malfunction (continued):

From the above discussion, it can be seen that at the time of original ISFSI design and construction that aircraft hazard was not considered a design basis event for the ISFSI due to it not being considered a design basis event for CCNPP. This in turn meant that it was not considered to be a malfunction initiator for the ISFSI which subsequently meant that any equipment important to safety would not be impacted and/or degraded.

With the above historical discussion now presented, the current aircraft hazard will be discussed. A very detailed aircraft hazards analysis (Reference 4) has been developed for the ISFSI in accordance with Section 3.5.1.6 of Reference 1. The Reference 4 analysis evaluates the following as directed by Section 3.5.1.6 of Reference 1:

1) Federal airways or airport approaches passing within 2 miles of the site.
2) All airports located within 5 miles of the site.
3) Airports with projected operations greater than $500d^2$ movements per year located within 10 miles of the site and greater than $1000d^2$ outside 10 miles, where d is the distance in miles from the site.
4) Military installations or any airspace usage that might present a hazard to the site. For some uses such as practice bombing ranges, it may be necessary to evaluate uses as far as 20 miles from the site.

There are eight airways situated in the vicinity of the ISFSI (References 2 & 3). Four (J14, J191, J61, and J37) are high altitude airways, and four (V31, V93, V16-157-213-229, and V20-33) are low altitude airways. References 2 & 3 show that only two of these eight airways (V31 and V93) meet the requirements for analysis stated in Section 3.5.1.6 of Reference 1 (i.e., the ISFSI either lies within the airway or is located less than two miles from one of the airway's outer borders). The other high and low altitude airways pass further than two miles from the ISFSI. The Reference 4 analysis determined that the total probability of an aircraft crash resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines, due to these airways, is 2.90×10^{-7} cr/yr. Reference 5 revisited this calculated probability and removed the "built-in" conservatism which in turn resulted in a revised probability of 5.45×10^{-8} cr/yr.

A helipad is located at the northern end of the site more than a 1,000 feet from the ISFSI. Generally, this helipad is used for corporate flights from BGE headquarters (Baltimore) and for an estimated six Medivac helicopter flights per year. Helicopter Transport Services, Inc., of Baltimore, MD, has indicated that the helicopter used to transport BGE personnel to and from the plant site is a Bell 206L helicopter weighing less than 3,000 pounds. This puts the helicopter in the NUREG/CR-5042 (Reference 6) category of "less than 12,000 pounds". The Medivac helicopter would also fall into the "less than 12,000 pounds" category. Table 6.4.2 of Reference 6 provides the probability of penetration of plant structures as a function of plant location, aircraft weight, and concrete thickness. Utilizing this table, knowing the ISFSI outer shell is composed of concrete at least three feet thick, the probability of a helicopter originating from an airport less than five miles from the ISFSI and penetrating the ISFSI is zero. Since the probability of penetration is zero, helicopter operations do not contribute to the overall total probability of aircraft accidents.

Besides the helipad, there is only one other air strip located within 5 miles of the ISFSI. The privately operated air strip, Mears Creek, is only sporadically used for leisure purposes by its owner/operator. Two small single-engine aircraft are based there and are the only aircraft that are expected to use the field. It can be reasonably assumed that these aircraft are not of the type that would approach 12,000 pounds in weight. For these reasons, the Mears Creek operations will not be considered any further in the overall total probability of aircraft accidents.

There are two airports (Chesapeake Ranch Airport and St. Mary's County Airport) which are located within ten miles of the ISFSI. Chesapeake Ranch Airport is approximately 6 miles southeast of the ISFSI. Flight traffic is

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199601328-001 **50.59 Log No.:** **72.48 Log No.: SE00004**

Probability of Malfunction (continued):

greatest during the summer with approximately six flights per week. Conservatively assuming this rate throughout the year would result in a total of slightly over 300 flights per year. For airports between five and ten miles from the ISFSI, the criterion of projected operations greater than $500d^2$ movements per year from Section 3.5.1.6 of Reference 1 can be calculated as $500 \times 6^2 = 18{,}000$ which is much greater than the estimate of 300 flights per year. Therefore, Chesapeake Ranch Airport will not be considered as a source of potential aircraft hazard. St. Mary's County Airport is approximately 10 miles southwest of the ISFSI with an estimated 3,400 flights per month, or 40,800 flights per year. Utilizing the above noted criterion of $500d^2$ results in $500 \times 10^2 = 50{,}000$ which is greater than the estimate of 40,800 flights per year. Therefore, St. Mary's County Airport will not be considered as a source of potential aircraft hazard.

Patuxent River Naval Air Station (Pax River NAS) is approximately 11 miles south of the ISFSI. There have been as many as 100,000 takeoffs and landings per year, though the projection for the next several years is 50,000 to 60,000 per year. The 100,000 flight figure is approximately equal to the number of flights that would be calculated as a screening criterion. therefore, Pax River NAS is considered to be a source of aircraft hazard.

The instrument approach landing and takeoff patterns for Pax River NAS are shown in References 7 & 8. It should be noted that, according to Patuxent River Air Operations, the exact flight paths shown in References 7 & 8 are used only in the event of loss of radar contact with the aircraft (and in training runs for such scenarios). Normally, the initial point for approach is at four miles from the air station, so approaches to Pax River NAS would, in most cases, remain seven miles from the ISFSI and plant site.

Three of the patterns (TACAN RWY 14, TACAN 1 RWY 24, and TACAN 1 RWY 32) displayed in References 7 & 8 approach the ISFSI and plant site. All of these are shown passing at a ten nautical mile radius from Pax River NAS, effectively flying planes directly overhead. Generally, planes shouldn't come any closer than 3 miles from the ISFSI since the Navy Airman's Information Manual directs pilots specifically to avoid flyovers of the CCNPP site. Pax River NAS Air Operations indicates that pilots are generally sent on three mile bypass loops around the CCNPP site to avoid such flyovers.

The TACAN RWY 14 approach depicted in Reference 8 is only used in sporadic training runs, as the normal initial point for overhead approach is four nautical miles out. The ten-mile radial pattern is only used (other than in training) if all radar contact with the aircraft is lost. The TACAN 1 RWY 24 and TACAN 1 RWY 32 ten mile radius patterns would be used only if there were a missed approach on a normal runway 24 or 32 landing and radar contact could not be maintained with the pilot of the aircraft. An actual Naval Facilities Engineering Command count of air traffic provided by Pax River NAS revealed that only 214 planes used these three routes in the past year. Utilizing the information discussed above, the Reference 4 analysis determined that the total probability of an aircraft crash resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines, due to Pax River NAS aircraft movement, is 8.72×10^{-9} cr/yr. Reference 5 revisited this calculated probability and utilized a more realistic military effective area along with a more reasonable probability of penetration which in turn resulted in a revised probability of 3.43×10^{-9} cr/yr.

Military usage of airspace in the vicinity of the site is generally covered by the activities at Pax River NAS and the military flights in local airways, both of which were previously mentioned above. Due to this and the lack of any other data suggesting otherwise, the Reference 4 analysis assumed that the overall rate for aircraft crashes due to military/other airspace usage was equal to 0 cr/yr. However, this is now known not to be true since military jet planes, which were determined to be from Andrews Air Force Base, were observed flying at a low altitude directly over the CCNPP site in December 1997. No exact data exists for this type of infrequent "random" non-airway type

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ACTIVITY: ES199601328-001 **50.59 Log No.:** **72.48 Log No.: SE00004**

Probability of Malfunction (continued):

of military flight. However, the potential hazard from this type of "random" non-airway type of military flight will be addressed later on in this "Probability of Malfunction" section.

The Department of Energy (DOE) conducts periodic radiation surveys over the plant site. As was noted on Page 3 of this Safety Evaluation, Table 6.4.2 of Reference 6 provides the probability of penetration of plant structures as a function of plant location, aircraft weight, and concrete thickness. Utilizing this table, knowing the ISFSI outer shell is composed of concrete at least three feet thick, the probability of the DOE helicopter penetrating the ISFSI is zero. Since the probability of penetration is zero, the DOE helicopter operations do not contribute to the overall total probability of aircraft accidents.

Without consideration of the "random" non-airway type of military flight, the total frequency of an aircraft crash resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is determined by summing the following:

• Aircraft crash frequency due to airways within 2 miles of the plant:	5.45×10^{-8} cr/yr.
• Aircraft crash frequency from airports within 5 miles of the site:	0 cr/yr.
• Aircraft crash frequency from Pax River NAS aircraft movement:	3.43×10^{-9} cr/yr.
• Aircraft crash frequency due to military/other airspace usage:	0 cr/yr.
• Aircraft crash frequency due to DOE radiation survey:	0 cr/yr.
Total crash frequency (probability)	5.79×10^{-8} cr/yr.

On Page 3.5.1.6-2 of NUREG-0800 (Standard Review Plan), Section 3.5.1.6 (Aircraft Hazards), which is utilized herein as a guideline (BGE is not committed to the Standard Review Plan), it states the following:

> "10 CFR Part 100, Section 100.10 as it relates to indicating that the site location, in conjunction with other considerations (such as plant design, construction, and operation), should insure a low risk of public exposure. This requirement is met if the probability of aircraft accidents resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is less than about 10^{-7} per year."

As noted above, the total probability of an aircraft crash resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is equal to 5.79×10^{-8} per year for the ISFSI, when ignoring "random" non-airway type of military flight, which is below the stated SRP level of acceptability of 1.0×10^{-7} per year.

The Reference 5 analysis looked at "random" non-airway flights occurring within various diameter circles utilizing the ISFSI as the center of the circle. A circle is utilized as the airway width since the aircraft could come from any direction.

Utilizing the following diameter circles, the number of "random" non-airway military flights that could occur, while still remaining below the SRP level of acceptability of 1.0×10^{-7} per year, are as follows:

• One mile circle	Number = 245/year
• One thousand foot circle	Number = 46/year

Though there is no existing data associated with the number of "random" non-airway military flights, general observations around the site conclude that it is apparent that flights directly over the ISFSI are relatively rare. It is unlikely that the number of actual "random" military flights significantly exceed the above stated values.

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199601328-001 **50.59 Log No.:** **72.48 Log No.: SE00004**

Probability of Malfunction (continued):

Therefore, the probability of an aircraft accident which could result in an offsite exposure level exceeding 10 CFR 100 limits is considered to be below the SRP level of acceptability of 1.0×10^{-7} per year.

From the above discussion on the current aircraft hazard for the ISFSI, it can be concluded that aircraft hazard is not a malfunction initiator since the probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is acceptably low. Therefore, it is concluded that any equipment important to safety will not be adversely impacted and/or degraded.

YES √ NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

In the above section it was shown that aircraft hazard does not have to be considered a design basis concern for the ISFSI since the calculated probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is considered to be below the SRP level of acceptability of 1.0×10^{-7} per year. Changes to aircraft flight patterns and/or probability has no affect on the design or method of operating equipment important to safety. Thus, it can be concluded that all equipment important to safety will operate as originally analyzed.

Based on the above, it is concluded that the current calculated aircraft hazard will not result in increased radiological consequences and will not increase the consequences of a malfunction of any equipment important to safety that has been previously evaluated in the SAR.

√ YES NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of an aircraft crash was not quantified during the timeframe of licensing and construction of the ISFSI. The existing aircraft hazard noted within the ISFSI USAR was derived from the CCNPP UFSAR where it was noted that aircraft from/to Pax River NAS would be no closer than approximately seven miles from the plant. As was noted on Page 2 of this safety evaluation (under the "Probability of Malfunction" section), the Directorate of Licensing at the U.S. Atomic Energy Commission concurred with BGE's conclusion that no special design provisions were required to be incorporated into CCNPP because the probability of an aircraft crash affecting the plant was acceptably low (implies a probability of less than 10^{-7}/year). Therefore, based on the CCNPP UFSAR the probability of an aircraft crash affecting the ISFSI was acceptably low at less than 10^{-7}/year.

In the above "Probability of Malfunction" section it was noted that the probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is below the SRP level of acceptability of 1.0×10^{-7} per year for the ISFSI. The probability of an aircraft accident during the timeframe of original construction and licensing of the ISFSI was never quantified. Since today's probability of an aircraft accident may be higher based on the fact that, at times, aircraft going into Pax River NAS fly practically overhead where previously they came no closer than seven miles from the ISFSI (as described in the USAR), the probability of occurrence of an accident will conservatively be considered to have increased. However, it should be noted that

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199601328-001 **50.59 Log No.:** **72.48 Log No.: SE00004**

Probability of Accident (continued):

the probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is considered to be below the SRP level of acceptability. Since the above probability of an aircraft accident is acceptably low, no additional design or procedural protection is required.

YES √ NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

Changes to the aircraft flight patterns and/or frequency (probability) have no affect on the design or method of operating equipment necessary to mitigate the consequences of previously analyzed accidents. As was noted above, the aircraft hazard is considered to be acceptably low and therefore no additional design or procedural protection is required for the ISFSI. Since the aircraft hazard is considered acceptably low (where additional design features are not required), it can be concluded that no action assumed to occur within the accident analysis of Chapter 8 will be degraded or prevented. Therefore, it is concluded that the current calculated aircraft hazard will not result in an increase of the Consequences of an Accident previously evaluated in the SAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

YES √ NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

All possible malfunctions have been previously analyzed. Aircraft hazard was addressed within the original design of the ISFSI. The frequency/probability of an aircraft crash was considered to be so low that special design provisions to protect against aircraft crashes did not have to be considered during construction of the ISFSI. The current calculated aircraft hazard is considered to be below the SRP level of acceptability of $1.0\text{x}10^{-7}$ per year. The possibility for a malfunction of a different type than any previously evaluated in the SAR is not created.

YES √ NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

As was noted above, aircraft accidents were considered within the original ISFSI design. The probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is still acceptably low and no special design provisions are required. Since an aircraft crash is not a design basis concern, it is not plausible that the possibility of a new accident is created which has not been previously evaluated in the SAR. There are also no new challenges to safety related equipment.

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199601328-001 **50.59 Log No.:** **72.48 Log No.: SE00004**

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

YES √ NO Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Bases:

CCNPP Unit 1 & 2 Technical Specifications
ISFSI Technical Specifications

Discussion of why the margin of safety is not reduced

The CCNPP and ISFSI Technical Specifications do not address or consider aircraft hazards for the ISFSI since the probability of an aircraft crash affecting the ISFSI, at the time of licensing and construction, was considered to be so low that no special design provisions were needed in the ISFSI for such an event. Since aircraft hazards did not have to be considered within the design of the ISFSI, no Margin of Safety was required or established for such a hazard. All of the assumptions stipulated within the Chapter 8 accident analysis would not be affected by such an event.

The calculated probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines, based on today's aircraft hazard, remains acceptably low and is considered to be below the SRP level of acceptability of 1.0×10^{-7} per year. Therefore, there is still no need for special design provisions within the ISFSI to guard against such an event. All of the assumptions stipulated within the Chapter 8 accident analysis remain unchanged. The ISFSI will continue to operate in such a manner that will ensure acceptable levels of protection for the health and safety of the public.

Complete for 72.48:

YES √ NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

As was noted previously, the probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is considered to be below the SRP level of acceptability of 1.0×10^{-7} per year. Therefore, since the requirements of 10 CFR Part 100 are maintained, it can be concluded that there will be no significant increase in occupational dose.

YES √ NO Will the proposed activity involve a significant unreviewed environmental impact?

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199601328-001 **50.59 Log No.:** **72.48 Log No.: SE00004**

A significant unreviewed environmental impact:

The aircraft hazard is an external event which will not create an environmental impact. As noted above, the frequency of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is considered to be below the SRP level of acceptability of 1.0×10^{-7} per year. Therefore, it can be concluded that the aircraft hazard does not create a significant unreviewed environmental impact.

References:

1) USNRC Regulatory Guide 1.70, Rev. 3, November 1978.
2) United States Government Flight Information Publication, IFR Enroute High Altitude - US Area H-6, January 1996.
3) United States Government Flight Information Publication, IFR Enroute Low Altitude - US Area L-28, January 1996.
4) NUS Calculation LA16.ISFSI Rev. 0 (BGE Calculation CA04039 Rev. 0), Aircraft Hazards Analysis for the Independent Spent Fuel Storage Installation.
5) Reliability Engineering Calculation 97-034 Rev. 3, IPEEE other External Event Analysis.
6) NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States", U.S. Nuclear Regulatory Commission, 1987.
7) Department of Defense Flight Information Publication, High Altitude United States Airport Diagrams - NE, April 1995.
8) Department of Defense Flight Information Publication, Low Altitude United States Airport Diagrams - VOL-10, May 1995.
9) Data Development Technical Support Document for the Aircraft Crash Risk Analysis Methodology (ACRAM) Standard (Draft). Lawrence Livermore National Laboratory, April 1995.
10) Summary of air traffic over the fix PXT, FAA Eastern Region, February 7, 1996. (also Attachment 4 under NUS Calculation LA16.AHA [BGE Calculation CA04040]).

Summary: (For NRC Report)

This activity, ESP ES199601328-001, revises the information currently provided within Revision 7 of the ISFSI USAR, under Section 2.2.1.1, on aircraft and their flight paths for Patuxent River Naval Air Station (Pax River NAS). The above noted section is outdated and does not reflect current conditions for aircraft utilizing Pax River NAS.

The actual aircraft hazard during original construction and licensing of the ISFSI was never quantified. This was due to the fact that the aircraft conditions were the same for both the ISFSI and CCNPP along with the fact that aircraft hazard for CCNPP (which was also never quantified) was judged to be acceptably low by the NRC at the time of construction and licensing of CCNPP. Section 3.1.2 of the Safety Evaluation by the Directorate of Licensing U.S. Atomic Energy Commission in the Matter of BGE CCNPP Units 1 & 2 dated 8/28/72 stated the following:

> "Considering the relatively small number of aircraft movements at these airports and their distances from the Calvert Cliffs site, the applicant concluded and we concur, that the probability of an aircraft crash affecting the plant is so low that no special design provisions should be made in the plant for such an event."

ATTACHMENT 3, SAFETY EVALUATION FORM

ACTIVITY: ES199601328-001 **50.59 Log No.:** **72.48 Log No.: SE00004**

Summary: (For NRC Report) [continued]

As part of CCNPP's Individual Plant Examination for External Events (IPEEE), a very detailed calculation was developed to address aircraft hazards for the ISFSI. This calculation addressed all of the hazards as directed by Section 3.5.1.6 of Regulatory Guide 1.70 (Reference 1) such as airways (V31 and V93) within 2 miles of the ISFSI, airports (the helipad at CCNPP and the Mears Creek air strip) within 5 miles of the ISFSI, airports (Chesapeake Ranch Airport and St. Mary's County Airport) within 10 miles of the ISFSI, Pax River NAS aircraft movement, and military/other airspace usage that might present a hazard to the ISFSI. Also, the Reference 5 calculation considered the hazard from the radiation survey that the DOE performs by flying a helicopter over the plant site several times. The results of this calculation (Reference 4) along with the Reference 5 calculation (which removed the "built-in conservatism within the Reference 4 calculation) determined that, when ignoring "random" non-airway type of military flight, the total probability of an aircraft crash resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is equal to 5.79×10^{-8} crash/year for the ISFSI. When considering "random" non-airway types of military flight and utilizing the following diameter circles, the number of "random" non-airway military flights that could occur, while still remaining below the SRP level of acceptability of 1.0×10^{-7} per year, are as follows:

• One mile circle	Number = 245/year
• One thousand foot circle	Number = 46/year

Section 3.5.1.6 of the SRP states the following:

> "10 CFR Part 100, Section 100.10 as it relates to indicating that the site location, in conjunction with other considerations (such as plant design, construction, and operation), should insure a low risk of public exposure. This requirement is met if the probability of aircraft accidents resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is less than about 10^{-7} per year."

The above noted calculated probability of 5.79×10^{-8} per year along with the above noted number of allowed "random" non-airway type of military flight, meets the above stated criteria of less than about 10^{-7} per year.

From the above discussion it becomes apparent that the probability of an accident may have increased. Though the probability of an accident may have increased, the risk that an aircraft crash would result in an offsite exposure level exceeding 10 CFR Part 100 limits is considered to be below the level of acceptability (i.e., 10^{-7} per year). Since aircraft hazard conditions have changed to the point that, at times, aircraft fly directly overhead versus seven miles from the ISFSI, as was originally described within the ISFSI SAR, it is being conservatively concluded that the probability of an accident has increased (the probability of an aircraft hazard was not previously quantified). Therefore, this activity will be considered to constitute a Unreviewed Safety Question and requires a review from the NRC.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Spacer Disk & Support Rod Material **72.48 Log No.: SE00005**

D3-DSC-21; 2/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 6

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 10·20·97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: R. O. Hardies Work Group: ME&IU	Resp. Indv.: R. H. Beall Work Group: NFM
10/20/97	10/21/97	10/20/97
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: K.C. ANSTEE K.C. Anstee
INDEPENDENT REVIEWER

Date 10-22-1997

[✓ Approved] Disapproved

Signature: for Michael J. Gahan
[✓ GS-DES] GS-TES, or PE-PDSU

Date 10·23·97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-126 Date: 11·3·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-3-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11/4/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Spacer Disk & Support Rod Material			**72.48 Log No.: SE00005**	
D3-DSC-21; 2/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 2 of 6**

Proposed Activity: To reconcile one identified difference between the NRC Safety Evaluation Report (SER) and the BGE Independent Spent Fuel Storage Installation (ISFSI) Updated Safety Analysis Report (USAR). This particular safety evaluation addresses the material used for the Dry Shielded Canister (DSC) spacer disks and support rods.

Reason for Activity: The NRC SER states that all DSC structural components are fabricated from type 304 stainless steel. The ISFSI USAR also states that all DSC structural components are fabricated from type 304 stainless steel, except the spacer disks and support rods may be fabricated from aluminum coated carbon steel. BGE requested an alternative material for the spacer disks and support rods to reduce fabrication costs. BGE approved this design change for construction in 1991. The ISFSI license was issued in November of 1992, and ISFSI loading operations began in November of 1993. All fifteen fuel loadings to date have been successful, of which seven of the DSCs were constructed with aluminum coated carbon steel spacer disks and support rods. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Special Note: This proposed activity was presented as a 10 CFR 72.48 safety evaluation to the Plant Operations and Safety Review Committee (POSRC) on April 6, 1992, Meeting No. 92-035. POSRC reviewed and recommended approval of the safety evaluation to the Plant General Manager, who subsequently approved the safety evaluation. Since this safety evaluation was approved prior to the issuance of the ISFSI 10 CFR 72.48 license, the change was incorporated in the first revision of the original SAR. As stated above, this safety evaluation was performed even though the change was incorporated into the ISFSI USAR. Seven of the fifteen DSC's loaded to date have aluminum coated carbon steel spacer disks and support rods.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are three major components of the NUHOMS-24P system that are addressed in this safety evaluation. Those three components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); and 3) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those three components.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and act as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

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Function(s) of affected SSC (con't):

Horizontal Storage Module (HSM) - each HSM is a reinforced, concrete structure constructed in place at the ISFSI site. Calvert Cliffs employs a 2 x 6 array, a massive concrete structure which consists of twelve HSM's in two rows of six. The side walls and roof are three feet thick, whereas the front walls are three and one half feet thick. There are two foot thick interior walls which separate each HSM and provide neutron and gamma shielding and prevent scatter in adjacent modules during DSC loading. The function of the HSM is to safely provide interim storage of the DSC's. The HSM provides the necessary radiological protection to the public at all times. Each HSM has been designed for worst case postulated and hypothetical accidents, including scenarios such as design basis tornadoes and tornado missiles.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.4, 3.6, 4.2, 5.1, 7.4, 8.1, and 8.2.

<u>Complete for 50.59 and 72.48:</u>

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

<u>Probability of Malfunction:</u>

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the USAR allowing the DSC spacer disks and support rods to be fabricated from type 304 stainless steel or aluminum coated carbon steel. The NRC SER currently states that all DSC structural components are fabricated from type 304 stainless steel. BGE requested the aluminum coated carbon steel as an alternative material for the spacer disks and support rods to reduce fabrication costs back in 1991 (The resultant savings per DSC was $10,500). The alternative material was evaluated by Pacific Nuclear Fuel Services in 1991 via vendor calculation no. BGE001.0216 (Carbon Steel DSC Basket Assembly) and concluded that it was structurally acceptable, and that the previous DSC structural vendor calculation no. BGE001.0203 (DSC Structural Analysis) was still valid. The calculation evaluated the DSC for allowable stresses, ductility, and corrosion resistance. The strength of carbon steel for structural support of the stored spent fuel exceeds that of the stainless steel.

The DSC basket assembly is constructed to ASME Boiler & Pressure Vessel Code, Division 1, Section NF (Component Supports). The original DSC's use stainless steel components (ASME SA-240, type 304). The newer DSC's have carbon steel support rods (ASME SA-696, Gr. B) and carbon steel spacer disks (ASME SA-516, Gr. 70).

As stated earlier, seven of the fifteen DSC's loaded to date have aluminum coated carbon steel spacer disks and support rods. All fifteen fuel moves to date have resulted in a smooth transfer of the DSC to the HSM.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

<u>Consequences of Malfunction:</u>

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the USAR allowing the DSC spacer disks and support rods to be fabricated from type 304 stainless steel or aluminum coated carbon steel. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Spacer Disk & Support Rod Material			72.48 Log No.: SE00005	
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NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of this proposed activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. Since the accident analysis was performed after the 1991 design change, it included the use of either type 304 stainless steel or aluminum coated carbon steel spacer disks and support rods. The USAR states that an actual drop event is not credible. The accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the use of either material was considered in the analysis, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. One possible malfunction of the DSC which is not described or evaluated in the USAR is the corrosion of the DSC carbon steel spacer disks and support rods due to exposure to spent fuel pool environment of borated water. The material corrosion properties are only relevant during transfer of fuel to the DSC in the spent fuel pool since the storage atmosphere is made inert with Helium and there is no oxygen present to support corrosion of the carbon steel spacer disks and support rods. To prevent any possible corrosion, cathodic protection was provided to all exposed carbon steel surfaces with a minimum 0.003 inches of flame sprayed aluminum coating. This not only protects the carbon steel during fuel loading, but also provides an additional corrosion barrier during long term storage. Aluminum corrosion rates in PWR water have been reported for immersed 3000 ppm boron water environment. These rates are insignificant, however, in that the Calvert Cliffs DSC's, under normal loading conditions, are exposed to the borated water for less than 48 hours. In addition, tests by Vectra Technologies concluded that no precipitates or corrosion products were visible in the test water and the water appeared clear. Chemical analysis of the water verified that aluminum released was less than 1 ppm. Therefore, the 0.003 inches of flame sprayed aluminum coating will remain in place and corrosion of the carbon steel will not take place.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. One accident scenario not described in the USAR is a chemical, galvanic, or other reaction in the DSC that could cause an ignition event. This relates to NRC Bulletin 96-04: Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks. This bulletin was the result of a hydrogen gas ignition event that occurred during the welding of the shield lid on a spent fuel storage cask at Wisconsin Electric Power Company's Point Beach Nuclear Plant on May 28, 1996. At Point Beach, an investigation concluded that the event occurred as a result of interaction between the borated spent fuel pool water and the zinc paint that coated the interior of the carbon steel canister inside the cask. The source of the hydrogen was the oxidation of zinc when it came in contact with the borated water.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Spacer Disk & Support Rod Material			**72.48 Log No.: SE00005**	
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The Calvert Cliffs DSC's are constructed entirely of type 304 stainless steel, except the spacer disks and support rods are fabricated from type 304 stainless steel or aluminum coated carbon steel. The BGE response to the bulletin addressed the flame sprayed aluminum coating on the carbon steel spacer disks and support rods, and the precautionary measures adopted by Calvert Cliffs. The next few paragraphs address the Calvert Cliffs response to NRC Bulletin 96-04 and the precautionary measures. The NRC acknowledged in an April 8, 1997 letter to Mr. C. H. Cruse, that it did not have a safety issue at that time regarding the NUHOMS-24P system.

It is well known that aluminum coatings on carbon steel react in aqueous media due to a combination of the galvanic corrosion and general corrosion methods. Since the aluminum coating is less noble than the carbon steel to which it is bonded, it will be subject to galvanic corrosion and function like a sacrificial coating. The contribution of radiolysis to the build-up of hydrogen in the DSC air space is minor compared to the contribution from corrosion. When hydrogen is generated by the simultaneous reaction of radiolysis and corrosion within the same water inventory, the combined generation of hydrogen will be suppressed due to competition for reaction products. Three sources of information were available to determine hydrogen generation for the Calvert Cliffs DSC's. They were laboratory testing, Duke Power measurements at Oconee, and computer simulation. For normal loading operations, the total elapsed time from the placement of the DSC top shield plug to the point at which the DSC cover plate is completely welded in place is expected to be less than 24 hours at temperatures ranging from about 70°F to 120°F. It was concluded that corrosion, coupled with radiolysis analysis results, indicate that the maximum hydrogen concentration is predicted to be 1.82%, which is less than half of the lower flammability limit of 4% hydrogen in air. Vectra Technologies has recommended that hydrogen monitoring should be performed with an alarm setpoint of 2.4%.

Based on the above, precautionary measures were adopted by Calvert Cliffs and incorporated into two procedures, ISFSI-01, "ISFSI Loading," and ISFSI-02, "ISFSI Unloading." The following steps have been added as a precautionary measure during ISFSI loading and unloading operations:

1) The DSC cavity will always be vented prior to welding of the inner lid during the loading operation, and prior to removing the inner lid during the unloading operation.
2) For operations involving DSC containing carbon steel coated with flame-sprayed aluminum, sampling for flammable gases will be performed. During ISFSI loading operation (ISFSI-01), sampling for flammable gases will be performed before any welding of the inner lid is complete and passes the dye penetrate test. If at any time the measured concentration of flammable gases inside the DSC rises above 50% of the flammability limit (which equates to an alarm setpoint of 2%), welding will stop and a purge of the DSC air space will begin. During the unloading operation (ISFSI-02), a continuous sampling of the DSC cavity will be performed while removing the inner lid. As in the case of the loading operation, if the measured concentration of flammable gases inside the DSC rises above 50% of the flammability limit (which equates to an alarm setpoint of 2%), the inner lid removal process will be stopped, and the DSC air space will be purged.

In summary, the possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

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ISFSI - DSC Spacer Disk & Support Rod Material			**72.48 Log No.: SE00005**	
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Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The design change provided an alternative material for the spacer disks and support rods to reduce fabrication costs. BGE approved this design change for construction in 1991. The change in material does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To reconcile one identified difference between the NRC Safety Evaluation Report (SER) and the BGE Independent Spent Fuel Storage Installation (ISFSI) Updated Safety Analysis Report (USAR). This particular safety evaluation addresses the material used for the Dry Shielded Canister (DSC) spacer disks and support rods.

Reason for Activity: The NRC SER states that all DSC structural components are fabricated from type 304 stainless steel. The ISFSI USAR also states that all DSC structural components are fabricated from type 304 stainless steel, except the spacer disks and support rods may be fabricated from aluminum coated carbon steel. BGE requested an alternative material for the spacer disks and support rods to reduce fabrication costs. BGE approved this design change for construction in 1991. The ISFSI license was issued in November of 1992, and ISFSI loading operations began in November of 1993. All fifteen fuel loadings to date have been successful, of which seven of the DSCs were constructed with aluminum coated carbon steel spacer disks and support rods. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Special Note: This proposed activity was presented as a 10 CFR 72.48 safety evaluation to the Plant Operations and Safety Review Committee (POSRC) on April 6, 1992, Meeting No. 92-035. POSRC reviewed and recommended approval of the safety evaluation to the Plant General Manager, who subsequently approved the safety evaluation. Since this safety evaluation was approved prior to the issuance of the ISFSI 10 CFR 72.48 license, the change was incorporated in the first revision of the original SAR. As stated above, this safety evaluation was performed even though the change was incorporated into the ISFSI USAR. Seven of the fifteen DSC's loaded to date have aluminum coated carbon steel spacer disks and support rods.

Activity Summary: After a thorough and intense review, it has been concluded that the proposed activity:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Annulus Fill Water **72.48 Log No.: SE00006**

A-DSC-2; 3/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11.7.97

YES	Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye	Resp. Indv.: C. J. Dobry	Resp. Indv.: R. H. Beall
Work Group: Licensing	Work Group: PES	Work Group: NFM
Getachew Tesfaye . 11/12/97	11/10/97	R.H. Beall 11/11/97
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11.19.97

Recommend Approval ✓ Recommend Disapproval _______ Signature: _______ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved _______ Signature: _______ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes _______ No X

Signature: _______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: __________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Annulus Fill Water				**72.48 Log No.: SE00006**
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Proposed Activity: To reconcile one identified difference between the NRC Safety Evaluation Report (SER) and the BGE Independent Spent Fuel Storage Installation (ISFSI) Updated Safety Analysis Report (USAR). This particular safety evaluation addresses the fill water for the DSC-TC annulus.

Reason for Activity: The SER states in one section that the Dry Shielded Canister (DSC)-Transfer Cask (TC) annulus is filled with borated water, and in another section states it is filled with demineralized water. The USAR states that the DSC-TC annulus is filled with demineralized water.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 4.2, 4.3, 4.4, 5.1, 7.2, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Annulus Fill Water			72.48 Log No.: SE00006	
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Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the USAR allowing the annulus between the DSC and cask to be filled with demineralized water and sealed with an inflatable seal. The purpose of this design has been to prevent contamination of the DSC outer surface by the spent fuel pool water.

The NRC SER states in Section 1.5.5 that the DSC-TC annulus is filled with borated water rather than demineralized water. However, Table 1-2, states in part that the water in the TC-DSC annulus is demineralized. The use of demineralized water is consistent with the manufacturer design as detailed in the NUHOMS-24P Topical Report, Section 5.1, Operation Description, which describes filling of the DSC-TC annulus with clean, demineralized water. The annulus between the DSC and cask is filled with demineralized water and sealed with an inflatable seal to prevent contamination of the DSC outer surface by the spent fuel pool water. Dry shielded canister loading procedures require that the annulus between the transfer cask and DSC be filled with demineralized water and sealed prior to immersion in the spent fuel pool.

This Safety Evaluation clarifies an existing condition and does not change the original design or operation of the DSC-TC annulus. This clarification has no detrimental impact on equipment important to safety. Therefore, this clarification will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the loading operation of the DSC while in the Spent Fuel Pool.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Annulus Fill Water **72.48 Log No.: SE00006**

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2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The use of demineralized water is consistent with the manufacturer design as detailed in the NUHOMS-24P Topical Report, Section 5.1, Operation Description, which describes filling of the DSC-TC annulus with clean, demineralized water. The annulus between the DSC and cask is filled with demineralized water and sealed with an inflatable seal to prevent contamination of the DSC outer surface by the spent fuel pool water. Dry shielded canister loading procedures require that the annulus between the transfer cask and DSC be filled with demineralized water and sealed prior to immersion in the spent fuel pool.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
3/4.2.3	This Technical Specification addresses the maximum allowable DSC Exterior Surface Contamination limits. The USAR requires filling the DSC-TC annulus with demineralized water, placing a mechanical seal over the annulus, and utilizing procedures which require examination of the annulus surfaces for smearable contamination. Therefore, there is no possibility of significant radionuclide release from the DSC exterior surface during transfer or storage.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. During transfer of the sealed DSC and subsequent storage in the HSM, the only postulated mechanism for the release of airborne radioactive material is the dispersion of non-fixed surface contamination on the DSC exterior. By filling the cask/DSC annulus with demineralized water, placing a mechanical seal over the annulus, and utilizing procedures which require examination of the annulus surfaces for smearable contamination, the contamination limits on the DSC can be kept below the permissible level for storage or transfer of fuel. Therefore, there is no possibility of significant radionuclide release from the DSC exterior surface during transfer or storage.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Annulus Fill Water				72.48 Log No.: SE00006
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Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To reconcile one identified difference between the NRC Safety Evaluation Report (SER) and the BGE Independent Spent Fuel Storage Installation (ISFSI) Updated Safety Analysis Report (USAR). This particular safety evaluation addresses the fill water for the DSC-TC annulus.

Reason for Activity: The SER states in one section that the Dry Shielded Canister (DSC)-Transfer Cask (TC) annulus is filled with borated water, and in another section states it is filled with demineralized water. The USAR states that the DSC-TC annulus is filled with demineralized water.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Helium Leak Test Seal Welds **72.48 Log No: SE00007**

A-DSC-3; 4/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO Involve an unreviewed safety question (USQ)?
NO Involve a change in the Technical Specifications/License Conditions or Bases?
NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO Involve a Significant Increase in Occupational Dose?
NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11·7·97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye	Resp. Indv.: C. J. Dobry	Resp. Indv.: R. H. Beall
Work Group: Licensing	Work Group: PES	Work Group: NFM
Getachew Tesfaye 11/12/97	11/10/97	10/11/97
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11·19·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: J R Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Helium Leak Test Seal Welds			72.48 Log No: SE00007	
A-DSC-3; 4/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To reconcile one identified difference between the NRC Safety Evaluation Report (SER) and the BGE Independent Spent Fuel Storage Installation (ISFSI) Updated Safety Analysis Report (USAR). This particular safety evaluation addresses when the helium leak test is performed on the seal welds for the DSC.

Reason for Activity: The NRC SER states to weld the DSC shield plug and then helium leak test the seal welds. This differs from the ISFSI USAR where the helium leak test is not performed at this point in the loading process.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. The four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 4.3, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Helium Leak Test Seal Welds **72.48 Log No: SE00007**

A-DSC-3; 4/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of performing the sequence for helium leak testing of the seal welds.

The NRC SER states in Section 1.5.5 to weld the DSC shield plug and then helium leak test the seal welds. However, BGE performs the following steps as detailed in the ISFSI USAR: 1) Seal weld top shield plug to DSC; 2) Perform NDE on seal weld; 3) Drain remaining water from DSC; 4) Vacuum dry DSC; 5) Backfill DSC with helium; 6) Perform helium leak test. Dye penetrant testing is performed upon completion of the seal weld. The reasoning behind this is to ensure the weld is in compliance with the BGE Weld Program, as it provides the primary closure for the DSC. In addition, the helium leak test would not be performed without the DSC vacuum dried. This order of operations is consistent with the manufacturer design as detailed in the NUHOMS-24P Topical Report, Section 5.1, Operation Description, which describes the performance of dye penetrant weld examination of the seal weld just after the weld is created.

This Safety Evaluation clarifies an existing condition and does not change the original design or operation of the DSC. This clarification has no detrimental impact on equipment important to safety. Therefore, this clarification will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the helium leak testing of the seal welds.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Helium Leak Test Seal Welds **72.48 Log No: SE00007**

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2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. BGE performs the following steps as detailed in the ISFSI USAR: 1) Seal weld top shield plug to DSC; 2) Perform NDE on seal weld; 3) Drain remaining water from DSC; 4) Vacuum dry DSC; 5) Backfill DSC with helium; 6) Perform helium leak test. Dye penetrant testing is performed upon completion of the seal weld. The reasoning behind this is to ensure the weld is in compliance with the BGE Weld Program, as it provides the primary closure for the DSC. In addition, the helium leak test would not be performed without the DSC vacuum dried. This order of operations is consistent with the manufacturer design as detailed in the NUHOMS-24P Topical Report, Section 5.1, Operation Description, which describes the performance of dye penetrant weld examination of the seal weld just after the weld is created.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
3/4.2.2	This technical specification addresses the minimum allowable leak tightness for DSC closure welds. To ensure compliance with this technical specification, the USAR specifies a certain sequence of events including the performance of NDE on the DSC seal welds prior to performance of helium leak testing. This order of operations is consistent with the manufacturer design as detailed in the NUHOMS-24P Topical Report, Section 5.1, Operation Description, which describes the performance of dye penetrant weld examination of the seal weld just after the weld is created. As such, the margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this activity. This activity responds to one identified difference between the NRC SER and the BGE ISFSI USAR. This activity clarifies an existing condition and does not change the original design or operation of the DSC. The clarification of the subject difference does not change any DSC component or function that would or could potentially increase occupational dose.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Helium Leak Test Seal Welds **72.48 Log No: SE00007**

A-DSC-3; 4/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 5 of 5**

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To reconcile one identified difference between the NRC Safety Evaluation Report (SER) and the BGE Independent Spent Fuel Storage Installation (ISFSI) Updated Safety Analysis Report (USAR). This particular safety evaluation addresses when the helium leak test is performed on the seal welds for the DSC.

Reason for Activity: The NRC SER states to weld the DSC shield plug and then helium leak test the seal welds. This differs from the ISFSI USAR where the helium leak test is not performed at this point in the loading process.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

SAFETY EVALUATION FORM

Page 1 of 10

ACTIVITY: ES 200100120	**50.59 Log No.: N/A**	**72.48 Log No.: SE00008**

ISFSI – Proof Pressure Testing of DSCs

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve an unreviewed safety question (USQ)?
☐ YES ☒ NO Involve a change in the Technical Specifications/License Conditions?
☒ YES ☐ NO Require a change or addition to the UFSAR/USAR/Technical Specification Bases?

Applicable to 10 CFR 72.48 Safety Evaluations

☐ YES ☒ NO Involve a Significant Increase in Occupational Dose?
☐ YES ☒ NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: B. H. Scott / M. Kaiseruddin
PRINTED NAME AND SIGNATURE
Department: DES-~~C~~MEU / Sargent & Lundy
Date: 02/28/01 02/28/01

☒ YES ☐ NO Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye **PRINTED NAME**	Resp. Ind.: R. H. Beall **PRINTED NAME**	Resp. Ind.: **PRINTED NAME**
SIGNATURE	**SIGNATURE**	**SIGNATURE**
Work Group: **Licensing**	Work Group: **NFM**	Work Group:
Date: 03/27/01	Date: 3/27/01	Date:

Approved ☒ Disapproved ☐	Approved ☒ Disapproved ☐
Signature (J. E. Remeniuk) **INDEPENDENT REVIEWER**	Signature P. D. PATEL Pravin Patel **GS-DES, GS-TSES, or PE-PDSU**
Date 4.9.01	Date 4/10/01

The POSRC has reviewed this evaluation according to NS-2-101.
POSRC Meeting No.: 01-028 Date: 4/11/01
Recommend Approval ☑ Recommend Disapproval ☐ Signature **POSRC CHAIRMAN** Date 4/11/01

Approved ☑ Disapproved ☐ Signature **PLANT GENERAL MANAGER** Date 4/11/01

The OSSRC has reviewed this evaluation according to NS-2-100.
Full OSSRC Committee review required? ☐ YES ☒ NO
Signature: Date: 7/12/01
OSSRC SES Chairman
If yes, OSSRC Meeting No. ______

Proposed Activity:

The proposed activity consists of making changes to the ISFSI USAR [Refs. 1 and 2]. The changes are being made to incorporate a description of the alternate way of leak testing that was performed on the first ten DSCs that were put in service. The DSCs impacted by this activity are BGE24P-R002, -R007, and -R010 through -R017.

The proposed activity does not involve any hardware change.

The USAR change consists of inserting a new paragraph in Section 3.3.2.1, as shown in Reference 2.

Background

The Independent Spent Fuel Storage Installation (ISFSI) at the Calvert Cliffs Nuclear Power Plant (CCNPP) utilizes the Nutech Horizontal Modular Storage (NUHOMS)-24P dry storage system. The system consists of concrete horizontal storage modules (HSMs), which provide passive storage for spent fuel assemblies that are placed within Dry Storage Canisters (DSCs). Twenty-four spent fuel assemblies are loaded into each DSC. Each DSC contains an outer leak-tight shell and an internal basket assembly. The outer shell provides the structural strength, shielding, and a leak-tight chamber for containing helium. The helium provides an inert atmosphere within the DSC.

The DSC shell is fabricated out of metal plate in a welded construction. Cylindrical portion of the shell contains girth and longitudinal welds. The bottom cover is welded to the shell near the bottom of the DSC. There is a circumferential weld near the top, which is made in the field after loading the fuel.

The NRC issued a Confirmatory Action Letter (CAL) [Ref. 6] to the DSC supplier, Vectra Technologies, in part to document the concern that leak testing was performed on DSCs in lieu of pressure testing in accordance with ASME B&PV Code, Section III, NB-6000. Vectra responded to the CAL, and committed to performing the pressure testing on DSCs, with the exception of those that were already loaded with spent fuel [Ref. 7]. Based on Vectra's response the NRC closed the CAL, with the clarification that "all in-service canisters should remain in service 'as is' without a NB-6000 proof-pressure test" [Ref. 8]. It is noted here that the DSCs impacted by this activity were loaded with fuel prior to issuance of the CAL.

This activity describes the approach CCNPP is taking to resolve the concern related to the lack of pressure testing for the ten in-service DSCs at CCNPP.

Analyses / Justifications

NRC's Safety Evaluation Report (SER) [Ref. 4] states about DSC leak testing that:

- The leak test performed during fabrication be a proof pressure tests in accordance with NB-6000,
- The leak test performed at the plant for assuring a gas tight seal for the top welds be helium leak detection which is very sensitive, and
- The leak test performed during fabrication for the bottom welds be a soap bubble film test per ANSI N14.5-1987.

ISFSI Tech Spec 3.2.2.2 also requires that the top weld be tested by the helium leak rate method. The Calvert Cliffs ISFSI License, Condition 16, seems to imply that the bottom weld shall also be tested by the helium leak test, which is in contradiction with the statement in the SER. A license amendment request has been submitted to the NRC to revise License Condition 16 so as to remove the discrepancy [Ref. 9].

The leak test requirements are essentially the same for NUHOMS general license. Vectra Technologies has summarized the requirements as follows [Ref. 7]:

- The NRC does not expect a NB-6000 proof pressure test of the DSC top and bottom closure welds either in the fabrication shop or in the field. (Per the CCNPP ISFSI SER and Tech Specs, a helium leak rate test is required for the top weld, and a soap bubble film test is required for the bottom welds.)
- The NRC does expect a NB-6000 proof pressure test of DSC shell hoop and longitudinal welds.

Vectra Technologies, in their response to the CAL [Ref. 7], covered not only the "general license" canisters but also others governed by 10 CFR 72 site licenses, such as those in use at CCNPP. This fact was acknowledged by the NRC in the attachment to their letter of 2/15/97 [Ref. 8]. Vectra argued that NB-6000 proof-pressure test for the in-service canisters was not necessary to demonstrate DSC's containment capability based on the following facts:

- The joining plates were sound.
- The weldments were sound because they used qualified materials, procedures, and welders. Also, the welds were made by a multi-pass process which effectively eliminated pin-hole leaks that might occur in a single-pass process.
- The shell material was very forgiving.
- The weldments were both surface and volumetrically examined (liquid penetrant test (PT) and radiograph test (RT)).
- The weldments were leak tested per ANSI N14.5.
- The pressure loading in a DSC was very low (unlike traditional pressure vessels, mechanical loads govern the DSC shell stresses, not the internal pressure).

The leak testing performed on the in-service DSCs was as follows: The bottom weld and the girth and longitudinal welds were tested by the soap bubble film test, and the top weld was tested by the helium leak test. Therefore the only welds not tested per the CCNPP ISFSI SER are the girth and longitudinal welds. CCNPP subsequently tested over 26 DSCs per NB-6000 with no canister failing the test [Refs. 9 and 10]. The fuel assemblies themselves were also tested before being loaded into the DSCs to ensure that there were no cladding failures [Ref. 11].

Vectra concluded that NB-6000 proof-pressure testing of the in-service DSCs was not practical, and that they should be accepted "as is". The NRC agreed with Vectra's conclusion [Ref. 8], and explained their reason for the agreement as follows. "The objective of the NB-6000 test is to demonstrate DSC's structural capability to maintain containment pressure boundary. Compared to the mechanical loads, such as cask impact, that govern the sizing of the DSC shell plate thickness and design of fabrication details to ensure adequate performance, the design internal pressure as a basis for an NB-6000 pressure test will generate a stress condition far less severe than is intended to demonstrate DSC's structural capability."

The facts provided by Vectra and the reason for acceptance provided by the NRC, as listed above, are true and applicable to the DSCs in use at CCNPP. Therefore, the in-service DSCs at the CCNPP are acceptable "as is".

Reason for Activity:

The activity is being performed partly to help close out the Issue Report IR0-037-091 [Ref. 3]. Proof pressure testing of the DSC girth and longitudinal welds was not done per the CCNPP ISFSI SER, to demonstrate the leak tightness. Leak tightness of the DSC is required to assure that the helium from the

DSC does not completely leak out over the storage period, which could otherwise expose the fuel cladding to potentially corrosive environment.

Function(s) of affected SSCs:

The affected SSCs are the DSCs.

The DSC is classified as important-to-safety per 10 CFR 72. It consists of an outer canister and an internal basket assembly. The sub-components of the internal basket assembly include the Spacer Discs, Support Rods, and Guide Sleeves. The internal basket assembly components are not attached structurally to the outer canister.

The DSC provides containment, shielding, criticality control, configuration control related to fuel retrievability, structural support, and thermal safety functions during loading operations, transfer operations, and storage. It is designed to remain intact under all accident conditions identified in the ISFSI USAR with no loss of function. Specific design functions of the DSC include the following:

1. Confinement - The DSC design provides mechanical confinement of the stored fuel assemblies to prevent the dispersion of particulate or gaseous radionuclides from the fuel. The primary function of the DSC is to provide confinement of the spent nuclear fuel. This is achieved by the stainless steel shell and two inner cover plates (top and bottom ends) which are welded to the shell assembly. There are also outer cover plates (top and bottom) to further assure containment integrity. The DSC confinement boundary is designed also to retain helium cover gas around the fuel in order to prevent corrosion of the fuel cladding and formation of expansive oxides in the fuel during storage.
2. Criticality Control - The DSC design provides for sub-criticality during the wet loading, DSC drying, and interim storage operations. This is accomplished by a combination of mechanical separation of the fuel assemblies by the internal basket assembly and neutron absorption in the steel guide sleeve material.
3. Fuel Support and Configuration Control - The DSC internal basket assembly provides support for the spent fuel assemblies during normal operations. The DSC also provides configuration control related to post accident recovery of spent nuclear fuel. The DSC is designed so that the worst-case postulated accidents, including a cask drop, will not result in deformation of the Internal Basket Assembly or the DSC shell to such a degree that retrieval of intact fuel assemblies is not assured.
4. Shielding - The DSC materials provide gamma radiation shielding. The DSC provides gamma shielding at its ends by the use of lead shield plugs. These provide ALARA dose rates at the top of the canister during drying and sealing operations and at the bottom for minimizing dose rates during DSC loading into the Horizontal Storage Module (HSM) and at the HSM door during storage.
5. Thermal - Decay heat is removed by thermal radiation and conduction from the DSC to the TC, and by thermal radiation and conduction and convection from the DSC to the HSM. The DSC maintains the helium cover gas, which is required for corrosion control. This cover gas improves the thermal performance of the DSC.

The functions of the internal basket assembly components are as follows:

6. Guide Sleeves – The guide sleeves establish storage compartments for 24 spent fuel assemblies within the DSC. The tops of the guide sleeves are flared to assist fuel-handling operators in guiding the spent fuel assemblies into the sleeves.
7. Spacer Discs – The spacer discs work together with the guide sleeves to maintain geometric separation of the fuel assemblies. The spacer discs support the weight of the guide sleeves, support rods and the spent nuclear fuel when the DSC is in a horizontal orientation.

8. Support Rods – The support rods maintain the spacer disk locations along the length of the DSC. They carry the weight of the guide sleeves and the spacer discs when the DSC is in a vertical orientation.

ISFSI USAR Revision No.: 9

ISFSI USAR Sections Reviewed:

The main chapters reviewed were 3, 4, 5, 7, and 8. The key Sections reviewed are listed as follows:

3.3.2	Protection by Multiple Confinement Barriers and Systems
4.2.1.2	Dry Shielded Canister (Structural Specifications)
4.2.3.2	Dry Shielded Canister Description
5.1.1.2	Fuel Loading
8.1.1.2	Dry Shielded Canister Analysis
8.1.1.3	Dry Shielded Canister Internal Basket Analysis
8.2.3.2	Accident Analysis
8.2.5	Cask Drop

Table 3.6-3	Summary of Design Criteria for Accident Conditions
Table 8.2-1	NUHOMS-24P Accident Loading Identification
Table 8.2-6	Maximum Dry Shielded Canister Stresses for Drop Accident Loads

Tech Spec Bases Amendment/Rev No.: 2

Calvert Cliffs Independent Spent Fuel Storage Installation Technical Specifications, Appendix A to Materials License No. SNM-2505, Amendment 2, June 30, 2000

Tech Spec Bases Reviewed:

3/4.2.2 DSC Closure Welds

CCNPP ISFSI SER
Section 2.2.3.2

Complete for 50.59 and 72.48:

1. **The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.**

☐ YES ☒ NO **May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Probability of Malfunction:

The proposed activity consists of making a change to the ISFSI USAR. The change is being made to incorporate a description of the leak testing which was performed on the first ten DSCs that were put into service. The type of leak testing that was performed was different from that stated by the NRC in the SER, which was the ASME B&PV Code, Section III, NB-6000 pressure test. However, the NRC accepted the in-service DSCs "as-is", and provided their reason for the acceptance as follows. "The objective of the NB-6000 test is to demonstrate DSC's structural capability to maintain containment pressure boundary. Compared to the mechanical loads, such as cask impact, that govern the sizing of the DSC shell plate thickness and design of fabrication details to ensure adequate performance, the design internal pressure as a basis for an NB-6000 pressure test will generate a stress condition far less severe than is intended to demonstrate DSC's structural capability."

The proposed activity does not involve any hardware changes.

Therefore, the probability of malfunction of equipment important to safety will not be increased because of the proposed changes.

☐ YES ☒ NO **May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?**

Consequences of Malfunction:

The malfunctions to be considered are those of the ISFSI important-to-safety components that are impacted by this activity, namely the DSCs.

The consequences of failure of the DSC are all related to the release of radioactivity into the atmosphere or the dose to operators or the public. The shielding and containment properties of the DSC are not compromised. For the NUHOMS-24P system, the NRC has accepted the use of in-service DSCs "as is", without requiring additional pressure testing. Therefore, the consequences of failure of the DSC will not be impacted by this activity.

☐ YES ☒ NO **May the probability of occurrence of an accident previously evaluated in the SAR be increased?**

Probability of Accident:

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the SAR. They consist of loss of shielding, external missiles, earthquake, flood, cask drop, lightning, blockage of air inlets and outlets, DSC leakage, DSC overpressurization, and forest fire.

There is no change to the design or operation of the NUHOMS system caused by this activity. This activity does not modify the external configuration of the DSC envelope. The interface between the DSC and the HSM during ISFSI operations and interim storage of the DSC remains unaffected. Therefore, the probability of occurrence of an accident involving loss of HSM air outlet shielding, or blockage of HSM air inlets and outlets will not increase.

Pressurization of the DSC due to fuel cladding failure is an accident scenario identified in USAR Section 8.2.9. The limiting DSC pressurization accident event is a rupture of fuel cladding together with blockage of the HSM vents. This activity does not compromise the fuel cladding, or the fuel rod integrity, to cause an increase in the probability of this accident.

DSC leakage is an accident scenario described in USAR Section 8.2.8. The USAR indicates that there are no credible events that would initiate this type of accident. As stated in the preceding paragraphs, the probability of an accident that would lead to cladding failure is not increased by this activity. This activity does not affect the design of the DSC pressure boundary. In fact, the USAR accident assumes that the fission products are released directly to the atmosphere instantaneously, which is a far greater leak rate than the one demonstrated through DSC leak testing. Therefore, the probability of DSC leakage is not increased.

☐ YES ☒ NO **May the consequences of an accident previously evaluated in the SAR be increased?**

Consequences of Accident:

The proposed activity consists of the USAR changes related to leak testing of the first ten DSCs that were loaded with the spent fuel.

The consequences of the cask drop accident on the DSC are described in the USAR. The accident does not lead to cladding rupture, or increased leakage of the fission products from the fuel.

The DSC leakage accident also would not result in any higher release of radioactivity, because the USAR accident assumes that the fission products are released directly to the atmosphere instantaneously, which is a far greater leak rate than the one demonstrated through DSC leak testing.

Therefore, consequences of an accident previously evaluated in the SAR will not be increased.

2. **The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.**

☐ YES ☒ NO **May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?**

Possibility of New Malfunction:

The proposed activity makes changes to the USAR related to leak testing of the first ten DSCs. None of the changes impact the environment, functioning, or the procedures related to the equipment important to safety. DSC leakage has been considered, therefore, there is no possibility created of a new malfunction in any of the important-to-safety ISFSI components.

ACTIVITY: ES 200100120	50.59 Log No.: N/A	72.48 Log No.: SE00008

ISFSI – Proof Pressure Testing of DSCs

☐ YES ☒ NO **May the possibility of an accident of a different type than any previously evaluated in the SAR be created?**

Possibility of New Accident:

Credible accidents analyzed for the Calvert Cliffs ISFSI are discussed in Section 8.2 of the USAR, and have been discussed previously. Evaluation of the proposed changes to the USAR showed that the important-to-safety components of ISFSI would maintain their safety functions. Since there is no change to the design or operation of the NUHOMS system caused by this activity, the possibility of an accident of a different type than any previously evaluated in the SAR would not be created.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any Technical Specification is not reduced.

☐ YES ☒ NO **Will the margin of safety as defined in the basis for any Technical Specification be reduced?**

Tech Spec Bases: 3.2.2

Discussion of why the margin of safety is not reduced:

The margin of safety is defined as the range of values between the acceptance limit reviewed and approved by the NRC as part of the licensing basis and the failure point [Ref. 17]. USAR Sections 3.2.5 and 3.3.2 define the acceptance criteria for ISFSI components, none of which would be exceeded. Therefore, the margin of safety would not be reduced.

Complete for 72.48:

☐ YES ☒ NO **Will the proposed activity involve a significant increase in occupational dose?**

A significant increase in occupational dose:

The radiation protection design and operation of the NUHOMS-24P dry cask storage system would not be changed by this proposed activity. The DSC would maintain the radioactivity confinement boundary. Because none of these attributes would be changed, the occupational doses summarized in USAR Table 7.4-1 would not be affected by this activity.

☐ YES ☒ NO **Will the proposed activity involve a significant unreviewed environmental impact?**

A significant unreviewed environmental impact:

The NUHOMS-24P dry cask storage system confinement and radiological shielding functions would not be reduced by this activity.

This activity would not affect any area of the plant site previously undisturbed for the ISFSI, and would not cause any reason for revision to the ISFSI Updated Environmental Report. This activity would not affect the environmental conditions associated with the ISFSI. Therefore, this activity would not involve an unreviewed environmental impact.

ISFSI – Proof Pressure Testing of DSCs

References:

1. Calvert Cliffs Independent Spent Fuel Storage Installation USAR, Rev. 9
2. CCNPP ISFSI USAR Change Request, UCR-00219
3. CCNPP Issue Report No. IR0-037-091, 8/27/95
4. Safety Evaluation Report (SER) for Baltimore Gas and Electric Company's Safety Analysis Report for an Independent Spent Fuel Storage Installation at Calvert Cliffs, November 1992
5. Technical Specifications for Calvert Cliffs ISFSI, Amendment 2
6. Nuclear Regulatory Commission's Confirmatory Action Letter Issued to Vectra Technologies, Inc., 7/7/1995
7. Vectra Technologies' Letter to the NRC, No. VF-95-047, Vectra Supplemental Response to Confirmatory Action Letter, 9/22/1995
8. NRC's Letter to Vectra Technologies, Clarification of Item 3 of Confirmatory Action Letter Dated July 7, 1995, 2/25/1997
9. BG&E Letter from Bruce Tracey to R. A. Ayers of Vectra Technologies, Quality Assurance Surveillance of Vectra Technologies, Inc. Fuel Services, July 23, 1996
10. BG&E Letter from Bruce Tracey to R. A. Ayers of Vectra Technologies, Quality Assurance Surveillance of Vectra Technologies, Inc. Fuel Services (NTE), August 26, 1996
11. Siemens Nuclear Power Reports EMF-92-146(P), Rev. 0, and EMF-94-136(P)
12. CCNPP Letter the NRC, License Amendment Request: Revision to License Conditions 9,12, and 16, November 16, 2000
13. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1983
14. RANOR, Inc., Procedure P-LTP-1, Rev. 0, Leak Testing Procedure, 1/22/91
15. Topical Report for the NUTECH Horizontal Modular Storage (NUHOMS) System for Irradiated Nuclear Fuel, NUH-002, Rev. 2
16. Calvert Cliffs Updated Final Safety Analysis Report (UFSAR), Rev. 26
17. NEI 96-07, Rev. 0, Guidelines for 10 CFR 50.59 Safety Evaluations, 09/97
18. Calvert Cliffs ISFSI Updated Environmental Report, Rev. 1

ACTIVITY: ES 200100120	**50.59 Log No.: N/A**	**72.48 Log No.: SE00008**

ISFSI – Proof Pressure Testing of DSCs

Summary: (For NRC Report, provide a brief overview)

Proposed Activity:

The proposed activity consists of making changes to the ISFSI USAR. The changes are being made to incorporate a description of the alternate way of leak testing which was used for the ten DSCs that were put in service first. The DSCs impacted by this activity are BGE24P-R002, -R007, and -R010 through -R017. The proposed activity does not involve any hardware changes.

Reason for Activity:

Proof pressure testing of the DSC girth and longitudinal welds was not done per ASME B&PV Code, Section III, NB-6000, as stated in the CCNPP ISFSI SER, to demonstrate the leak tightness. Leak tightness of the DSC is required to assure that the helium from the DSC does not completely leak out over the storage period, which could otherwise expose the fuel cladding to potentially corrosive environment.

Activity Summary:

The USAR change being made documents the following. The only welds on the in-service DSCs, which were not pressure-tested per the CCNPP ISFSI SER were the girth and longitudinal welds; instead they were tested by the soap bubble film test. The soap bubble film test performed on those welds measures the air leakage.

Continued use of those DSCs "as is" is justified based on the facts that the plate and weld materials and welding procedures used were sound, weldments were both surface and volumetrically examined, weldments were leak tested per ANSI N14.5, and the pressure loading in a DSC was very low.

CCNPP subsequently tested over 26 DSCs per NB-6000 with no canister failing the test. The fuel assemblies themselves were also tested before being loaded into the DSCs to ensure that there were no cladding failures.

NB-6000 proof-pressure testing of the in-service DSCs is not practical, and based on the above facts, they should be accepted "as is". The NRC agreed with this conclusion for the general license canisters, as well those governed by 10 CFR 72 site-specific licenses, such as those in use at CCNPP, and provided their reason for the agreement as follows. "The objective of the NB-6000 test is to demonstrate DSC's structural capability to maintain containment pressure boundary. Compared to the mechanical loads, such as cask impact, that govern the sizing of the DSC shell plate thickness and design of fabrication details to ensure adequate performance, the design internal pressure as a basis for an NB-6000 pressure test will generate a stress condition far less severe than is intended to demonstrate DSC's structural capability."

USQ Determination: This activity was evaluated against the criteria of 10CFR72.48(a)(2), such as the probability of occurrence or the consequences of an accident or the malfunction of equipment important to safety, and it was concluded that it does not involve an unreviewed safety question (USQ).

ES200100120-000

ATTACHMENT 3, POSRC/PRC PRESENTATION FORM

POSRC/PRC PRESENTATION FORM

Presentation Date: April 11, 2001

Presenter: Mohammed Kaiseruddin

Extension: 4120

Procedures or Activity:

10CFR72.48 Safety Evaluation, Log No. SE00008, ISFSI – Proof Pressure Testing of DSCs

Purpose of Presentation:

☒ Recommendation for Approval

☐ Information

☐ Close OI

☐ Extend OI

Activity Summary: (See POSRC/PRC Presenter's Guide III.A.1):

The proposed activity consists of making changes to the ISFSI USAR. The changes are being made to incorporate a description of the alternate method of leak testing which was used for the ten DSCs that were put in service first, instead of ASME Section III, NB-6000 proof pressure-testing as stated in the CCNPP ISFSI SER. The DSCs impacted by this activity are BGE24P-R002, -R007, and -R010 through -R017. The proposed activity does not involve any hardware changes.

Continued use of those DSCs "as is" is justified based on the facts that the plate and weld materials and welding procedures used were sound, weldments were both surface and volumetrically examined, weldments were leak tested per ANSI N14.5, and the pressure loading in a DSC was very low.

CCNPP subsequently tested over 26 DSCs per NB-6000 with no canister failing the test. The fuel assemblies themselves were also tested before being loaded into the DSCs to ensure that there were no cladding failures.

NB-6000 proof-pressure testing of the in-service DSCs is not practical, and based on the above facts, they should be accepted "as is". The NRC concurred with this conclusion for the general license canisters, as well those governed by 10 CFR 72 site-specific licenses such as those in use at CCNPP. The USAR change consists of inserting a new paragraph in Section 3.3.2.1.

Safety Issues Involved: (See POSRC/PRC Presenter's Guide II.B, C, D, E, and III.A.2):

The affected systems, structures and components (SSCs) are DSCs BGE24P-R002, -R007, and -R010 through -R017.

The DSC is classified as important-to-safety per 10 CFR 72. It provides containment, shielding, criticality control, configuration control related to fuel retrievability, structural support, and thermal safety functions during loading operations, transfer operations, and storage. It is designed and tested to assure that it contains helium, thus preserving a non-corrosive environment for fuel cladding.

Recommendations to POSRC or PRC: (See POSRC/PRC Presenter's Guide II.F, G. H, and III.A.3 and F): Recommend approval of this 10CFR72.48 safety evaluation.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Closure Operations **72.48 Log No.: SE00009**

A-DSC-8; 6/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk PRINTED NAME AND SIGNATURE Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	CJ Dobry 11/10/97 SIGNATURE / DATE	RH Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER

Date 11/13/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for [✓ GS-DES], GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11-19-97

Recommend Approval ✓ Recommend Disapproval _______ Signature: JL Call Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved _______ Signature: JL Call Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes _______ No X

Signature: JR Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ________________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Closure Operations **72.48 Log No.: SE00009**

A-DSC-8; 6/129; ES199601368 Supplement 001 Revision 0000 Page 2 of 5

Proposed Activity: To reconcile one identified difference between the NRC Safety Evaluation Report (SER) and the BGE Independent Spent Fuel Storage Installation (ISFSI) Updated Safety Analysis Report (USAR). This safety evaluation addresses a difference in regard to filling the TC-DSC (Transfer Cask-Dry Shielded Canister) annulus area during transfer DSC closure operations.

Reason for Activity:. The SER identifies the difference in use of water in the TC-DSC (Transfer Cask-Dry Shielded Canister) annulus between the NUHOMS-24P System (Nutech Horizontal Modular Storage) defined in the TR (Topical Report) and the Calvert Cliffs SAR without acknowledging the fact that Calvert Cliffs allows varying the sequence of operations detailed in Chapter 5 of the ISFSI USAR, as long as the limiting conditions for operation are not exceeded.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections Reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.4, 4.2, 4.3, 4.4, 5.1, 7.4, 8.1, and 8.2, including figure 5.1-1, "ISFSI Loading Operations Flowchart.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Closure Operations			**72.48 Log No.: SE00009**	
A-DSC-8; 6/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 3 of 5**

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the USAR varying the sequence of DSC closure operations. The NRC SER states that the water in the DSC/cask annular gap will be drained when the water inside the DSC is drained following completion of the top shield primary seal weld, and that subsequent DSC closure operations will be performed with the DSC cavity and the annular gap dry. The shielding calculations were performed assuming that water would be present in the annular gap when the DSC is flooded, and that the annular gap would be drained when the DSC is drained. The ISFSI USAR provides in Section 5.1.1 a narrative that describes operations unique to the Nutech Horizontal Modular Storage (NUHOMS) systems, such as draining, drying and closure of the dry shielded canister (DSC), in some detail but it is not intended to be limiting or restrictive. Operational procedures may be revised according to the requirements of the plant, provided that the limiting conditions of operation are not exceeded. The justification is that over time, procedures will be revised to incorporate more efficient and/or safer work practices. BGE has written and revised technical procedure ISFSI-01, Independent Spent Fuel Storage Installation (ISFSI) Loading. The procedure requires that demineralized water remain in the annulus through the last closure operation for ALARA purposes. This approach is conservative, in that shielding is provided for as long as possible.

This Safety Evaluation clarifies an existing condition and does not change the original design or operation of the DSC. This clarification has no detrimental impact on equipment important to safety. Therefore, this clarification will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. None of the accident scenarios address the DSC closure operations..

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this activity. As stated above, there are no possible accidents of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Closure Operations			**72.48 Log No.: SE00009**	
A-DSC-8; 6/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 4 of 5**

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The ISFSI USAR provides in Section 5.1.1 a narrative that describes operations unique to the Nutech Horizontal Modular Storage (NUHOMS) systems, such as draining, drying and closure of the dry shielded canister (DSC), in some detail but it is not intended to be limiting or restrictive. Operational procedures may be revised according to the requirements of the plant, provided that the limiting conditions of operation are not exceeded. The justification is that over time, procedures will be revised to incorporate more efficient and/or safer work practices. BGE has written and revised technical procedure ISFSI-01, Independent Spent Fuel Storage Installation (ISFSI) Loading. The procedure requires that demineralized water remain in the annulus through the last closure operation for ALARA purposes. This approach is conservative, in that shielding is provided for as long as possible.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
3/4.2.3	This Technical Specification addresses the maximum allowable DSC Exterior Surface Contamination limits. The USAR requires filling the DSC-TC annulus with demineralized water, placing a mechanical seal over the annulus, and utilizing procedures which require examination of the annulus surfaces for smearable contamination. In addition, technical procedure ISFSI-01, Independent Spent Fuel Storage Installation (ISFSI) Loading, requires that demineralized water remain in the annulus through the last closure operation for ALARA purposes. This approach is conservative, in that shielding is provided for as long as possible. Therefore, there is no possibility of significant radionuclide release from the DSC exterior surface during transfer or storage.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this activity. Since technical procedure ISFSI-01, Independent Spent Fuel Storage Installation (ISFSI) Loading, requires that demineralized water remain in the annulus through the last closure operation for ALARA purposes, shielding is provided for as long as possible.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Closure Operations **72.48 Log No.: SE00009**

A-DSC-8; 6/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To reconcile one identified difference between the NRC Safety Evaluation Report (SER) and the BGE Independent Spent Fuel Storage Installation (ISFSI) Updated Safety Analysis Report (USAR). This safety evaluation addresses a difference in regard to filling the TC-DSC (Transfer Cask-Dry Shielded Canister) annulus area during transfer DSC closure operations.

Reason for Activity: The SER identifies the difference in use of water in the TC-DSC (Transfer Cask-Dry Shielded Canister) annulus between the NUHOMS-24P System (Nutech Horizontal Modular Storage) defined in the TR (Topical Report) and the Calvert Cliffs SAR without acknowledging the fact that Calvert Cliffs allows varying the sequence of operations detailed in Chapter 5 of the ISFSI USAR, as long as the limiting conditions for operation are not exceeded.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Guide Sleeve Corner Weld Design Change **72.48 Log No.: SE00010**

D3-DSC-1; 7/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye	Resp. Indv.: C. J. Dobry	Resp. Indv.: R. H. Beall
Work Group: Licensing	Work Group: PES	Work Group: NFM
	11/10/97	R.H. Beall 11/11/97
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A Wright M.A. WRIGHT
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11-19-97

Recommend Approval ✓ Recommend Disapproval ____ Signature: ____ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ____ Signature: ____ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ____ No X

Signature: ____ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.:____

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Guide Sleeve Corner Weld Design Change			**72.48 Log No.: SE00010**	
D3-DSC-1; 7/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 2 of 5**

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) guide sleeve corner weld.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Guide Sleeve Corner Weld Design Change **72.48 Log No.: SE00010**

D3-DSC-1; 7/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the guide sleeve corner weld design change. The subject guide sleeve corner weld design change meets the weld design requirements as established by Pacific Nuclear Fuel Services (PNFS). This change does not affect any design or licensing requirements. The original weld on the drawing was a full length (100%) fillet weld. The revised weld is an intermittent weld which provides approximately 30% of the length of the original weld. However, because the fuel loads are transmitted directly to the spacer discs, the weld stresses are negligible, and the full length weld was not necessary. Intermittent welding is a common practice for components not subjected to direct loading. The weld symbol on the drawing indicates that the 4" continuous weld is required at both ends. This is to ensure that the free ends are not unwelded. In addition, Note 12 on the drawing (84-002-E) states that the welds shall be ground flush outside and shall not protrude inside the guide sleeve. This is required to protect the fuel assemblies from protruding weld material. Based on this information, the subject design change will not affect the form, fit or function of the DSC guide sleeve, is not detrimental to the structural integrity of the guide sleeve, will not obstruct insertion of the fuel assemblies into the guide sleeves and will not adversely affect the ability of the DSC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the weld design change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Guide Sleeve Corner Weld Design Change			72.48 Log No.: SE00010	
D3-DSC-1; 7/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

has not changed as a result of the weld design change, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject guide sleeve corner weld design change meets the weld design requirements as established by Pacific Nuclear Fuel Services (PNFS). This change does not affect any design or licensing requirements. The original weld on the drawing was a full length (100%) fillet weld. The revised weld is an intermittent weld which provides approximately 30% of the length of the original weld. However, because the fuel loads are transmitted directly to the spacer discs, the weld stresses are negligible. Based on this information, the subject design change will not affect the form, fit or function of the DSC guide sleeve, is not detrimental to the structural integrity of the guide sleeve, will not obstruct insertion of the fuel assemblies into the guide sleeves and will not adversely affect the ability of the DSC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a guide sleeve corner weld design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The weld change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Guide Sleeve Corner Weld Design Change **72.48 Log No.: SE00010**

D3-DSC-1; 7/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) guide sleeve corner weld.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Guide Sleeve Tolerance Design Change **72.48 Log No.: SE00011**

D3-DSC-2; 8/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	CJ Dobry 11/10/97 SIGNATURE / DATE	R H Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER

Date 11/13/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for [✓ GS-DES], GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11.19.97

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: J R Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Guide Sleeve Tolerance Design Change			72.48 Log No.: SE00011	
D3-DSC-2; 8/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a tolerance design change to the DSC (Dry Shielded Canister) guide sleeve.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this report.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Guide Sleeve Tolerance Design Change			72.48 Log No.: SE00011	
D3-DSC-2; 8/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the guide sleeve tolerance design change. The subject change in tolerances meets the current design requirement as established by Pacific Nuclear Fuel Services (PNFS). These dimensions are not critical for proper DSC operation. This change has no effect on DSC design. The design change relaxed the tolerances for the lengths of the guide sleeve and flare from ± 0.06" to ± 0.12". The drawing (84-002-E) indicates that the tolerances are applied at the top end for the flare and overall length, and both are +/- 0.12". Since the spacer disc detail shows that the guide sleeves are separated by 1.50", the flare tolerance is acceptable. For the length, the possible additional 0.06" is negligible, and is therefore acceptable. The subject tolerance change will not affect the form, fit or function of the guide sleeve, and will not adversely affect the ability of the DSC to perform its intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the tolerance design change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the tolerance design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Guide Sleeve Tolerance Design Change			72.48 Log No.: SE00011	
D3-DSC-2; 8/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. The subject guide sleeve length and flare dimensional tolerance change meets the design requirements as established by Pacific Nuclear Fuel Services (PNFS). The design change relaxed the tolerances for the lengths of the guide sleeve and flare from ± 0.06" to ± 0.12". The drawing (84-002-E) indicates that the tolerances are applied at the top end for the flare and overall length, and both are +/- 0.12". Since the spacer disc detail shows that the guide sleeves are separated by 1.50", the flare tolerance is acceptable. For the length, the possible additional 0.06" is negligible, and is therefore acceptable. Based on this information, the subject tolerance change will not affect the form, fit or function of the guide sleeve, and will not adversely affect the ability of the DSC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a tolerance design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The tolerance change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Guide Sleeve Tolerance Design Change				**72.48 Log No.: SE00011**
D3-DSC-2; 8/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 5 of 5**

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a tolerance design change to the DSC (Dry Shielded Canister) guide sleeve.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Spacer Disc Surface Finish Requirements **72.48 Log No.: SE00012**
D3-DSC-3; 9/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11-7-97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright H.A. WRIGHT
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11-19-97

Recommend Approval ✓ Recommend Disapproval _______ Signature: _______ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved _______ Signature: _______ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes _______ No X

Signature: J R Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ____________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Spacer Disc Surface Finish Requirements	72.48 Log No.: SE00012
D3-DSC-3; 9/129; ES199601368 Supplement 001 Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the surface finish requirements of the DSC (Dry Shielded Canister) spacer disc interior cut-outs.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Spacer Disc Surface Finish Requirements			**72.48 Log No.: SE00012**	
D3-DSC-3; 9/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 3 of 5**

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the spacer disc surface finish requirements design change. The subject design change allowed the interior finish of the spacer disc cut-outs to be relaxed to 500 micro-inches to provide the fabricator a wider choice of cutting methods. The DSC spacer disc cut-out interior surface finish design change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The cut-out finish only needs to be adequate to allow the guide sleeves to be installed in the basket. The drawing (84-002-E) indicates that the outside dimension of a guide sleeve is (8.70" +/- 0.03") + 2(0.105" +/- 0.005") = maximum 8.95". The spacer disc cut-out 9.10" +/- 0.015", thus it has a minimum opening of 9.085". This leaves a gap of (0.135 / 2) = 0.0675" on each side of the guide sleeve (less the finish coat) when centered during insertion. The 500 micro-inch finish, which equals (500)(1/1,000,000) = 0.0005", is insignificant compared to 0.0675". The drawing symbol indicates that this is the minimum finish required. Even if a finish of, say 10 mils is applied, that is still only 0.01" thick". Therefore, the change to the 500 micro-inch surface finish is adequate to allow the guide sleeves to be installed in the basket. This change therefore does not affect the operation or design of the DSC. The subject change in surface finish will not affect the form, fit or function of the spacer disc, will not adversely affect the ability of the DSC to perform it's intended design function, and has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the surface finish requirement design change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Spacer Disc Surface Finish Requirements **72.48 Log No.: SE00012**

D3-DSC-3; 9/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the surface finish requirement design change, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. The subject design change allowed the interior finish of the spacer disc cut-outs to be relaxed to 500 micro-inches to provide the fabricator a wider choice of cutting methods. The DSC spacer disc cut-out interior surface finish design change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The cut-out finish only needs to be adequate to allow the guide sleeves to be installed in the basket. The 500 micro-inch finish is insignificant compared to the 0.0675" on each side of the guide sleeve when centered during insertion. Therefore, the change to the 500 micro-inch surface finish is adequate to allow the guide sleeves to be installed in the basket. This change therefore does not affect the operation or design of the DSC. The subject change in surface finish will not affect the form, fit or function of the spacer disc, will not adversely affect the ability of the DSC to perform it's intended design function, and has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a spacer disc surface finish requirements design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The finish requirements change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Spacer Disc Surface Finish Requirements			72.48 Log No.: SE00012	
D3-DSC-3; 9/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the surface finish requirements of the DSC (Dry Shielded Canister) spacer disc interior cut-outs.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Grapple Ring Material Classification Design Change **72.48 Log No.: SE00014**

D3-DSC-5; 11/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	R H Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN ~~III~~
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11-19-97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Grapple Ring Material Classification Design Change			72.48 Log No.: SE00014	
D3-DSC-5; 11/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) grapple ring.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Grapple Ring Material Classification Design Change **72.48 Log No.: SE00014**

D3-DSC-5; 11/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the grapple ring material classification design change. The subject activity changed the grapple ring material classification from ASTM A-240 Type 304 to ASME SA-240 Type 304 (see drawing 84-003-E). The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). The grapple ring material classification was upgraded for consistency with the grapple ring code classification. This change does not adversely affect the design, since the material did not change, only the classification of the material. Although the grapple ring material did not change, the designation was upgraded to ASME from ASTM. The ASME material has the same properties as the ASTM, but, in addition, material documentation (chemical/physical characteristics) would be provided. The subject material designation change does not affect the form, fit or function of the grapple ring, and will not adversely affect the ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the grapple ring material classification design change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the grapple ring material classification design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Grapple Ring Material Classification Design Change			72.48 Log No.: SE00014	
D3-DSC-5; 11/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity changed the grapple ring material classification from ASTM A-240 Type 304 to ASME SA-240 Type 304. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). The grapple ring material classification was upgraded for consistency with the grapple ring code classification. This change does not adversely affect the design, since the material did not change, only the classification of the material. Although the grapple ring material did not change, the designation was upgraded to ASME from ASTM. The ASME material has the same properties as the ASTM, but, in addition, material documentation (chemical/physical characteristics) would be provided. The subject material designation change does not affect the form, fit or function of the grapple ring, and will not adversely affect the ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a grapple ring material classification design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The grapple ring material classification change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Grapple Ring Material Classification Design Change			72.48 Log No.: SE00014	
D3-DSC-5; 11/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) grapple ring.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of Grapple Ring Inside Grinding Requirement **72.48 Log No.: SE00015**

D3-DSC-6; 12/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for ~~GS-DES~~, GS-TES, or PE-PDSU
MICHAEL J. GAHAN ~~III~~
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11-19-97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-18-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of Grapple Ring Inside Grinding Requirement			72.48 Log No.: SE00015	
D3-DSC-6; 12/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) grapple ring.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of Grapple Ring Inside Grinding Requirement			72.48 Log No.: SE00015	
D3-DSC-6; 12/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the deletion of the grapple ring grinding requirement design change (see drawing 84-003-E). The subject design change deleted the grinding requirement from the inside surface of the grapple ring to facilitate fabrication (grinding of the surface is difficult) and is not required (a weld crown on the inside surface does not affect the operation of the grapple or DSC). The subject deletion of grapple ring inside surface grinding requirements meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The subject design change will not affect the form, fit or function of the grapple ring , and will not adversely affect the ability of the DSC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the grapple ring grinding requirement design change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the grapple ring grinding requirement design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of Grapple Ring Inside Grinding Requirement 72.48 Log No.: SE00015

D3-DSC-6; 12/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject design change deleted the grinding requirement from the inside surface of the grapple ring to facilitate fabrication (grinding of the surface is difficult) and is not required (a weld crown on the inside surface does not affect the operation of the grapple or DSC). The subject deletion of grapple ring inside surface grinding requirements meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The subject design change will not affect the form, fit or function of the grapple ring , and will not adversely affect the ability of the DSC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a grapple ring grinding requirement design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The grapple ring grinding requirement design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of Grapple Ring Inside Grinding Requirement			72.48 Log No.: SE00015	
D3-DSC-6; 12/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) grapple ring.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top & Bottom Shield Plug Tolerance Design Change **72.48 Log No.: SE00016**
D3-DSC-7; 13/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11-7-97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	R H Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A. Wright
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11.19.97

Recommend Approval ✓ Recommend Disapproval ______ Signature: John T Carroll Date 11-18-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: John T Carroll Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No ______

Signature: ______________________ Date: ____________
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ____________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top & Bottom Shield Plug Tolerance Design Change			72.48 Log No.: SE00016	
D3-DSC-7; 13/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) top and bottom shield plug plate thickness tolerances.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top & Bottom Shield Plug Tolerance Design Change			72.48 Log No.: SE00016	
D3-DSC-7; 13/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the top and bottom shield plug tolerance design change. The subject design change broadened the thickness tolerances of the top and bottom shield plug plates to provide maximum / minimum calculated thicknesses (see drawing 84-003-E). The subject change in tolerances meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The material thickness in the shield plugs were re-dimensioned to clarify the minimum and maximum acceptable thicknesses of each material. The thicknesses shown represent the bounding analyzed configurations of the DSC. The thickness requirements were computed during the DSC structural analysis. The DSC end plugs provide confinement and radiation shielding. The bottom end plug sandwiches lead between an outer plate and an inner plate of Type 304 stainless steel. The top plug is formed by two covers, separately welded to the DSC stainless steel shell. The inner cover and outer cover are manufactured from Type 304 stainless steel with lead placed between these cover plates. The increase in DSC weight due to the increase in the shield plug thickness is negligible as compared to the weight of the entire DSC. The subject tolerance change will not affect the form, fit or function of the top and bottom shield plugs, and will not adversely affect the ability of the DSC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the top and bottom shield plug tolerance design change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top & Bottom Shield Plug Tolerance Design Change 72.48 Log No.: SE00016

D3-DSC-7; 13/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the top and bottom shield plug tolerance design change, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. The subject design change broadened the thickness tolerances of the top and bottom shield plug plates to provide maximum / minimum calculated thicknesses. The subject change in tolerances meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The material thickness in the shield plugs were re-dimensioned to clarify the minimum and maximum acceptable thicknesses of each material. The thicknesses shown represent the bounding analyzed configurations of the DSC. The thickness requirements were computed during the DSC structural analysis. The increase in DSC weight due to the increase in the shield plug thickness is negligible as compared to the weight of the entire DSC. The subject tolerance change will not affect the form, fit or function of the top and bottom shield plugs, and will not adversely affect the ability of the DSC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a top and bottom shield plug tolerance design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The top and bottom shield plug tolerance design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top & Bottom Shield Plug Tolerance Design Change **72.48 Log No.: SE00016**

D3-DSC-7; 13/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 5 of 5**

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) top and bottom shield plug plate thickness tolerances.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of DSC Lead Casting Full Surface Requirement **72.48 Log No.: SE00017**

D3-DSC-8; 14/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO Involve an unreviewed safety question (USQ)?
NO Involve a change in the Technical Specifications/License Conditions or Bases?
NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO Involve a Significant Increase in Occupational Dose?
NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11·7·97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 **SIGNATURE / DATE**	11/10/97 **SIGNATURE / DATE**	11/11/97 **SIGNATURE / DATE**

[✓ Approved] Disapproved

Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER

Date 11/13/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for [✓ **GS-DES**], **GS-TES, or PE-PDSU**
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11·19·97

Recommend Approval ✓ Recommend Disapproval ________ Signature: John T Carroll Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: J R Lemon Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of DSC Lead Casting Full Surface Requirement			72.48 Log No.: SE00017	
D3-DSC-8; 14/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) lead shielding inspection requirement.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of DSC Lead Casting Full Surface Requirement	72.48 Log No.: SE00017
D3-DSC-8; 14/129; ES199601368 Supplement 001 Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the deletion of the lead casting full surface requirement design change. The subject design change deleted the requirement that the lead casting have full surface contact with the shield plug plates to facilitate the fabrication and pouring of the lead plugs (see drawing 84-003-E). The subject design change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). Full surface contact between the lead casting and the shield plug plates is neither necessary nor detectable, since any gap between the lead and the shell would not form a streaming path due to the geometry of the DSC. The gamma scan required by the fabrication specification ensures that full shielding thickness is obtained. This change therefore does not affect the design or operation of the DSC and does not impact any safety or licensing criteria.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the deletion of the lead casting full surface requirement design change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the deletion of the lead casting full surface requirement design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of DSC Lead Casting Full Surface Requirement			**72.48 Log No.: SE00017**	
D3-DSC-8; 14/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 4 of 5**

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject design change deleted the requirement that the lead casting have full surface contact with the shield plug plates to facilitate the fabrication and pouring of the lead plugs. The subject design change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). Full surface contact between the lead casting and the shield plug plates is neither necessary nor detectable, since any gap between the lead and the shell would not form a streaming path due to the geometry of the DSC. The gamma scan required by the fabrication specification ensures that full shielding thickness is obtained. This change therefore does not affect the design or operation of the DSC and does not impact any safety or licensing criteria.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. This activity involved the deletion of the lead casting full surface requirement. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The deletion of the lead casting full surface requirement design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1, since the gamma scan required by the fabrication specification ensured that full shielding thickness was obtained.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of DSC Lead Casting Full Surface Requirement **72.48 Log No.: SE00017**

D3-DSC-8; 14/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) lead shielding inspection requirement.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Shell Weld Preparation **72.48 Log No.: SE00018**

D3-DSC-9; 15/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO Involve an unreviewed safety question (USQ)?
NO Involve a change in the Technical Specifications/License Conditions or Bases?
NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO Involve a Significant Increase in Occupational Dose?
NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk PRINTED NAME AND SIGNATURE Department: NED-CEU 42-01-04 Date: 11·7·97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11·19·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ POSRC CHAIRMAN Date 11-19-97

Approved ✓ Disapproved ______ Signature: ______ PLANT GENERAL MANAGER Date 11-19-97

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ OSSRC SES CHAIRMAN Date: 1/30/98

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Shell Weld Preparation **72.48 Log No.: SE00018**

D3-DSC-9; 15/129; ES199601368 Supplement 001 Revision 0000 Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the inside surface of the DSC (Dry Shielded Canister) shell for the top cover bevel weld preparation.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation..

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Shell Weld Preparation **72.48 Log No.: SE00018**

D3-DSC-9; 15/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the design change to the inside surface of the DSC shell for the top cover weld preparation. The subject design change added a bevel of 0.75" x 22.5° to the inside surface of the DSC shell for the top cover weld preparation to facilitate DSC shell fabrication (see 84-003-E). The top end of the DSC shell has a tendency to bow inward during the placement of the shield plug weldment. This change prevents the movement of the shell from interfering with the installation of the top cover plate. The subject change in weld prep configuration meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS), and does not affect the in-use configuration of the DSC. The revising of the DSC shell inside surface weld prep configuration for installation of the top cover plate does not reduce the joint weld throat thickness and does not have a detrimental affect on the weld configuration strength. The subject change does not compromise design integrity, will not affect the form, fit or function of the DSC shell configuration, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the design change to the inside surface of the DSC shell for the top cover weld preparation does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Shell Weld Preparation **72.48 Log No.: SE00018**

D3-DSC-9; 15/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 4 of 5**

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the design change to the inside surface of the DSC shell for the top cover weld preparation, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject design change added a bevel of 0.75" x 22.5° to the inside surface of the DSC shell for the top cover weld preparation to facilitate DSC shell fabrication. The top end of the DSC shell has a tendency to bow inward during the placement of the shield plug weldment. This change prevents the movement of the shell from interfering with the installation of the top cover plate. The subject change in weld prep configuration meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS), and does not affect the in-use configuration of the DSC. The revising of the DSC shell inside surface weld prep configuration for installation of the top cover plate does not reduce the joint weld throat thickness and does not have a detrimental affect on the weld configuration strength. The subject change does not compromise design integrity, will not affect the form, fit or function of the DSC shell configuration, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a design change to the inside surface of the DSC shell for the top cover weld preparation. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The design change to the inside surface of the DSC shell for the top cover weld preparation does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Shell Weld Preparation			72.48 Log No.: SE00018	
D3-DSC-9; 15/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the inside surface of the DSC (Dry Shielded Canister) shell for the top cover bevel weld preparation.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top Cover Plate Weld Design Changes **72.48 Log No.: SE00019**

D3-DSC-10; 16/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk **PRINTED NAME AND SIGNATURE** Department: NED-CEU 42-01-04 Date: 11.7.97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye	Resp. Indv.: C. J. Dobry	Resp. Indv.: R. H. Beall
Work Group: Licensing	Work Group: PES	Work Group: NFM
Getachew Tesfaye 11/12/97	11/10/97	RH Beall 11/11/97
SIGNATURE / DATE	**SIGNATURE / DATE**	**SIGNATURE / DATE**

[✓ Approved] Disapproved

Signature: Mark A Wright M.A. WRIGHT
INDEPENDENT REVIEWER

Date 11/13/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for **GS-DES, GS-TES, or PE-PDSU**
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11.19.97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: JB Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top Cover Plate Weld Design Changes			**72.48 Log No.: SE00019**	
D3-DSC-10; 16/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 2 of 5**

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) top cover plate weld preparation and top cover to shell weldment.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top Cover Plate Weld Design Changes **72.48 Log No.: SE00019**

D3-DSC-10; 16/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the design change to the DSC top cover plate weld preparation and top cover to shell weldment. The subject design change revised the top cover plate weld preparation and the top cover to shell weldment. The top cover weld preparation was reduced from 45 degrees to 30 degrees, and the top cover plate to shell weldment was changed from a 5/8" J weld to a 5/8" V weld (see drawings 84-006-E and 84-009-E). The reason for this design change was to prevent burning through the plate during fabrication. The revised weld symbol, but unchanged plate details, give an identical weld throat to that of the original design. The subject change in weld configuration meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). This change has no effect on the DSC structural calculations. The subject design change does not affect the DSC shell to top cover plate weld NDE (Non-destructive examination) requirements, does not reduce the weld throat thickness, and does not have a detrimental affect on the weld strength. The subject change does not compromise design integrity, will not affect the form, fit or function of the top cover plate to DSC shell configuration, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the design change to the DSC top cover plate weld preparation and top cover to shell weldment does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top Cover Plate Weld Design Changes			72.48 Log No.: SE00019	
D3-DSC-10; 16/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the design change to the DSC top cover plate weld preparation and top cover to shell weldment, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. The subject design change revised the top cover plate weld preparation and the top cover to shell weldment. The top cover weld preparation was reduced from 45 degrees to 30 degrees, and the top cover plate to shell weldment was changed from a 5/8" J weld to a 5/8" V weld. The reason for this design change was to prevent burning through the plate during fabrication. The revised weld symbol, but unchanged plate details, give an identical weld throat to that of the original design. The subject change in weld configuration meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). This change has no effect on the DSC structural calculations. The subject design change does not affect the DSC shell to top cover plate weld NDE (Non-destructive examination) requirements, does not reduce the weld throat thickness, and does not have a detrimental affect on the weld strength. The subject change does not compromise design integrity, will not affect the form, fit or function of the top cover plate to DSC shell configuration, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity involved a design change to the DSC top cover plate weld preparation and top cover to shell weldment. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The design change to the DSC top cover plate weld preparation and top cover to shell weldment does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top Cover Plate Weld Design Changes **72.48 Log No.: SE00019**

D3-DSC-10; 16/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) top cover plate weld preparation and top cover to shell weldment.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Siphon Tube Dimensional Design Change **72.48 Log No.: SE00020**

D3-DSC-12; 17/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO Involve an unreviewed safety question (USQ)?
NO Involve a change in the Technical Specifications/License Conditions or Bases?
NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO Involve a Significant Increase in Occupational Dose?
NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11.7.97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A.Wright
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11.19.97

Recommend Approval ✓ Recommend Disapproval _______ Signature: _______ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved _______ Signature: _______ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes _______ No X

Signature: J R Lemons Date: 1/30/98
OSSRC/SES CHAIRMAN

If yes, OSSRC Meeting No.: ________________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Siphon Tube Dimensional Design Change			**72.48 Log No.: SE00020**	
D3-DSC-12; 17/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 2 of 5**

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) siphon tube.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Siphon Tube Dimensional Design Change			72.48 Log No.: SE00020	
D3-DSC-12; 17/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the siphon tube dimensional design change. The subject change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The siphon tube was previously dimensioned to be 0.12" below the face of the bottom cover. It is now dimensioned to be 0.19" +/- 0.06" (see drawing 84-004-E), which gives it the range of 0.13" to 0.25" above the bottom of the (bottom cover plate) cut out, which is 0.25" deep. The subject change in siphon tube dimensioning was made to better control the position of the siphon tube in order to reduce the likelihood of the tube becoming clogged during water removal. The siphon tube is used with the Vacuum Drying System to pump water from the canister to the spent fuel pool. The cut-out is designed to capture what little excess water will remain at the bottom of the canister that could not physically be removed. The fact that the siphon tube will be no higher than the top of the cut-out makes this change acceptable. This change does not affect the DSC design or operation, and will not have a detrimental impact on the water removal ability of the siphon tube, in fact, the water removal ability is enhanced. The subject change does not compromise design integrity, will not affect the form, fit or function of the siphon tube or DSC shell configuration, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the siphon tube dimensional design change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Siphon Tube Dimensional Design Change **72.48 Log No.: SE00020**
D3-DSC-12; 17/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the siphon tube dimensional design change, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject change in siphon tube dimensioning was made to better control the position of the siphon tube in order to reduce the likelihood of the tube becoming clogged during water removal. The siphon tube is used with the Vacuum Drying System to pump water from the canister to the spent fuel pool. The cut-out is designed to capture what little excess water will remain at the bottom of the canister that could not physically be removed. The fact that the siphon tube will be no higher than the top of the cut-out makes this change acceptable. This change does not affect the DSC design or operation, and will not have a detrimental impact on the water removal ability of the siphon tube, in fact, the water removal ability is enhanced. The subject change does not compromise design integrity, will not affect the form, fit or function of the siphon tube or DSC shell configuration, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a siphon tube dimensional design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The siphon tube dimensional design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Siphon Tube Dimensional Design Change			72.48 Log No.: SE00020	
D3-DSC-12; 17/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) siphon tube.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of Drain & Fill Block Bottom Weld **72.48 Log No.: SE00021**

D3-DSC-13; 18/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	R H Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Mark A Wright M.A. WRIGHT
INDEPENDENT REVIEWER

Date 11/13/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11-19-97

Recommend Approval ✓ Recommend Disapproved ______ Signature: ______ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of Drain & Fill Block Bottom Weld			72.48 Log No.: SE00021	
D3-DSC-13; 18/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) drain and fill block weldment to the DSC shell.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of Drain & Fill Block Bottom Weld			72.48 Log No.: SE00021	
D3-DSC-13; 18/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the deletion of the drain & fill block bottom weld design change. The subject design change deleted the weld between the bottom of the drain/fill block and the DSC shell . The weld was a 5/16" fillet weld, as originally found on DWG DUK-03-1003 of the NUHOMS TR (Topical Report). The subject design change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The function of the weld is served by the fillets on the side and the groove weld on top of the drain & fill block (see 84-004-E). This is structurally acceptable as there will be over 37 inches of weld for the drain & fill block. This change does not affect the DSC design or operation, does not compromise design integrity, will not affect the form, fit or function of the drain and fill block to DSC shell joint, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the deletion of the drain & fill block bottom weld design change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the deletion of the drain & fill block bottom weld design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of Drain & Fill Block Bottom Weld **72.48 Log No.: SE00021**

D3-DSC-13; 18/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject design change deleted the weld between the bottom of the drain/fill block and the DSC shell . The subject design change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The function of the weld is served by the fillets on the side and the groove weld on top of the drain & fill block. This is structurally acceptable as there will be over 37 inches of weld for the drain & fill block. This change does not affect the DSC design or operation, does not compromise design integrity, will not affect the form, fit or function of the drain and fill block to DSC shell joint, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity involved the deletion of the drain & fill block bottom weld. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The deletion of the drain & fill block bottom weld design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of Drain & Fill Block Bottom Weld **72.48 Log No.: SE00021**

D3-DSC-13; 18/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) drain and fill block weldment to the DSC shell.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Maximum Length Design Change **72.48 Log No.: SE00022**

D3-DSC-14; 19/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-for CEU 42-01-04 Date: 11·7·97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for [✓ GS-DES], GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11·19·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Maximum Length Design Change			**72.48 Log No.: SE00022**	
D3-DSC-14; 19/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 2 of 5**

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the length of the DSC (Dry Shielded Canister).

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.1, 3.3, 3.4, 3.6, 4.2, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Maximum Length Design Change **72.48 Log No.: SE00022**

D3-DSC-14; 19/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

<u>Complete for 50.59 and 72.48:</u>

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the DSC maximum length design change. The subject design change increased the DSC design length from 172.87" to 172.93" (see drawing 84-006-E). The subject change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). This change was made to better control this critical interface dimension. The DSC will fit inside the transfer cask under worst case thermal conditions, and as such, this design change has a negligible effect on the interface between the DSC and the transfer cask. The additions of 0.06" of material is negligible from a structural standpoint. The subject change does not compromise design integrity, will not affect the form, fit or function of the DSC, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the DSC maximum length design change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the DSC maximum length design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Maximum Length Design Change			72.48 Log No.: SE00022	
D3-DSC-14; 19/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject design change increased the DSC design length from 172.87" to 172.93". The subject change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). This change was made to better control this critical interface dimension. The DSC will fit inside the transfer cask under worst case thermal conditions, and as such, this design change has a negligible effect on the interface between the DSC and the transfer cask. The additions of 0.06" of material is negligible from a structural standpoint. The subject change does not compromise design integrity, will not affect the form, fit or function of the DSC, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a DSC maximum length design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The DSC maximum length design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Maximum Length Design Change **72.48 Log No.: SE00022**

D3-DSC-14; 19/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the length of the DSC (Dry Shielded Canister).

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Shield Plug Keyway Design Change **72.48 Log No.: SE00023**

D3-DSC-16; 20/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk PRINTED NAME AND SIGNATURE Department: NED-CEU 42-01-04 Date: 11·7·97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER

Date 11/13/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for GS-DES/GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11·19·97

Recommend Approval ✓ Recommend Disapproval ____ Signature: John T Carroll POSRC CHAIRMAN Date 11-19-97

Approved ✓ Disapproved ____ Signature: John T Carroll PLANT GENERAL MANAGER Date 11-19-97

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ____ No X

Signature: JR Lemons OSSRC SES CHAIRMAN Date: 1/30/98

If yes, OSSRC Meeting No.: ____

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Shield Plug Keyway Design Change **72.48 Log No.: SE00023**

D3-DSC-16; 20/129; ES199601368 Supplement 001 Revision 0000 Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) top lead plug side casing plate keyway.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the shield plug keyway design change. The subject design change permitted the use of a single bent plate to fabricate the keyway in the top shield plug in lieu of five plates joined by four double v-groove welds surrounding the drain & fill block (see drawing 84-007-E). The reason for this design change was to provide the fabricator the option to bend one piece of material as compared to welding five plates together. The subject change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The use of a single plate to form the shield plug keyway in place of several joined plated does not affect the DSC design or operation. The subject design change, providing the option to form the DSC shield plug keyway from one piece of material, will not adversely affect the form, fit or function of the DSC or the assembly interface between the top shield plug and drain & fill block. Additionally this design change will not have a detrimental impact on the DSC's ability to perform it's intended design function.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the shield plug keyway design change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the shield plug keyway design change, there will be no increase in the accident dose consequences already described in the USAR. The consequences of the accidents described in Chapter 8 of the ISFSI

ATTACHMENT 3, SAFETY EVALUATION FORM

USAR vary from none to minimal worker exposure. None of these accident scenario consequences will be impacted by the subject design change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject design change permitted the use of a single bent plate to fabricate the keyway in the top shield plug in lieu of five plates joined by four double v-groove welds surrounding the drain & fill block. The reason for this design change was to provide the fabricator the option to bend one piece of material as compared to welding five plates together. The subject change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The use of a single plate to form the shield plug keyway in place of several joined plated does not affect the DSC design or operation. The subject design change, providing the option to form the DSC shield plug keyway from one piece of material, will not adversely affect the form, fit or function of the DSC or the assembly interface between the top shield plug and drain & fill block. Additionally this design change will not have a detrimental impact on the DSC's ability to perform it's intended design function.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a shield plug keyway design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The shield plug keyway design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Shield Plug Keyway Design Change			72.48 Log No.: SE00023	
D3-DSC-16; 20/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) top lead plug side casing plate keyway.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Realignment of Top Cover Plate Lifting Holes **72.48 Log No.: SE00024**

D3-DSC-17; 21/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for [✓ GS-DES] GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11-19-97

Recommend Approval ✓ Recommend Disapproval ____ Signature: ____ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ____ Signature: ____ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ____ No X

Signature: ____ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ____

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Realignment of Top Cover Plate Lifting Holes			72.48 Log No.: SE00024	
D3-DSC-17; 21/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) top cover plate.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) as related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Realignment of Top Cover Plate Lifting Holes			72.48 Log No.: SE00024	
D3-DSC-17; 21/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the realignment of the top cover plate lifting holes. The subject design change realigned the top cover lifting holes to the same locations as those in the top shield plug to reduce streaming through the lifting holes (see drawings 84-002-E, 84-007-E and 84-009-E). The function of the top cover plate lifting holes is to assist with the lifting, positioning, and placement of the 1-1/4" thick top cover plate on the DSC. The lifting holes for both the top shield plug assembly and the top cover plate are right above the support rod locations. There was no change to the diameter, thread pitch, or hole depth. The subject design change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). This change does not affect the DSC interface with any other item, including the welding machine. In addition, this change does not affect the DSC design or operation. This design change has no detrimental impact on the DSC structure, and does not cause an interference with any other component (including the transfer cask). The subject change does not compromise design integrity, will not affect the form, fit or function of the DSC top cover plate, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the realignment of the top cover plate lifting holes does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Realignment of Top Cover Plate Lifting Holes 72.48 Log No.: SE00024

D3-DSC-17; 21/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the realignment of the top cover plate lifting holes, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject design change realigned the top cover lifting holes to the same locations as those in the top shield plug to reduce streaming through the lifting holes. The function of the top cover plate lifting holes is to assist with the lifting, positioning, and placement of the 1-1/4" thick top cover plate on the DSC. The lifting holes for both the top shield plug assembly and the top cover plate are right above the support rod locations. There was no change to the diameter, thread pitch, or hole depth. The subject design change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). This change does not affect the DSC interface with any other item, including the welding machine. In addition, this change does not affect the DSC design or operation. This design change has no detrimental impact on the DSC structure, and does not cause an interference with any other component (including the transfer cask). The subject change does not compromise design integrity, will not affect the form, fit or function of the DSC top cover plate, and will not adversely affect the DSC's ability to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a realignment of the top cover plate lifting holes. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The realignment of the top cover plate lifting holes does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Realignment of Top Cover Plate Lifting Holes **72.48 Log No.: SE00024**

D3-DSC-17; 21/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) top cover plate.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Addition of Shield Plug Backing Bar **72.48 Log No.: SE00025**

D3-DSC-18; 22/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A.Wright
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for [✓ GS-DES]/GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11.19.97

Recommend Approval ✓ Recommend Disapproval ____ Signature: ____ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ____ Signature: ____ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ____ No X

Signature: ____ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ____

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Addition of Shield Plug Backing Bar			72.48 Log No.: SE00025	
D3-DSC-18; 22/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) side casing to top casing plate joint configuration.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Addition of Shield Plug Backing Bar			72.48 Log No.: SE00025	
D3-DSC-18; 22/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the addition of the shield plug backing bar. The subject change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The bar is 1/2" x 1/2" ASTM A479 or ASTM A240 Type 304 and is non-safety related (see drawing 84-006-E). The reason for this change was to prevent the lead plug from "wicking" into the side casing plate to casing plate weld pool during fabrication. The joint between the side casing plate and the top casing plate is made after lead has been poured into the shield plug. Lead has a tendency to wick through the joint and into the weld pool during welding. A backing bar has been added in accordance with NB-4435 to reduce the likelihood of this occurrence (see drawing 84-007-E). The addition of the backing bar does not affect the structural calculations. The presence of the backing bar (and the corresponding lack of lead) will slightly increase dose rates during installation of the shield plug. This slight increase will have a negligible effect on occupational doses, which will be offset by the increased ease of placing the shield plug to shell weldment. The shorter time required to install the plug should offset the higher dose rate. Therefore, based on the above information, the subject change does not compromise design integrity, will not affect the form, fit or function of the DSC side casing plate to top casing plate joint configuration, and will not adversely affect the DSC's ability to perform it's intended design function. This design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the addition of the shield plug backing bar does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Addition of Shield Plug Backing Bar			72.48 Log No.: SE00025	
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NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the addition of the shield plug backing bar, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject change meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The bar is 1/2" x 1/2" ASTM A479 or ASTM A240 Type 304 and is non-safety related. The reason for this change was to prevent the lead plug from "wicking" into the side casing plate to casing plate weld pool during fabrication. The joint between the side casing plate and the top casing plate is made after lead has been poured into the shield plug. Lead has a tendency to wick through the joint and into the weld pool during welding. A backing bar has been added in accordance with NB-4435 to reduce the likelihood of this occurrence. The addition of the backing bar does not affect the structural calculations. The presence of the backing bar (and the corresponding lack of lead) will slightly increase dose rates during installation of the shield plug. This slight increase will have a negligible effect on occupational doses, which will be offset by the increased ease of placing the shield plug to shell weldment. The shorter time required to install the plug will offset the higher dose rate. Therefore, based on the above information, the subject change does not compromise design integrity, will not affect the form, fit or function of the DSC side casing plate to top casing plate joint configuration, and will not adversely affect the DSC's ability to perform it's intended design function. This design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Addition of Shield Plug Backing Bar			72.48 Log No.: SE00025	
D3-DSC-18; 22/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity involved the addition of the shield plug backing bar. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The presence of the backing bar (and the corresponding lack of lead) will slightly increase dose rates during installation of the shield plug. This slight increase will have a negligible effect on occupational doses, which will be offset by the increased ease of placing the shield plug to shell weldment. The shorter time required to install the plug will offset the higher dose rate. Therefore, the addition of the shield plug backing bar does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The subject design change does not involve the ISFSI Updated Environmental Report or deal with any environmental issues.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) side casing to top casing plate joint configuration.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top Shield Plug Casing Plate Thickness Tolerance Change 72.48 Log No.: SE00026

D3-DSC-19; 23/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11.7.97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A Wright M.A. WRIGHT
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11.19.97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top Shield Plug Casing Plate Thickness Tolerance Change **72.48 Log No.: SE00026**

D3-DSC-19; 23/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 2 of 5**

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) top shield plug casing plate thickness tolerances.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top Shield Plug Casing Plate Thickness Tolerance Change	72.48 Log No.: SE00026
D3-DSC-19; 23/129; ES199601368 Supplement 001 Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the top shield plug casing plate thickness tolerance change. The subject design change allowed the thickness of the top shield plug top casing plate to vary between 0.24" and 0.52" to allow the fabricator flexibility in machining the top shield plug casing plate (see drawing 84-007-E). The previously allowed range was 0.24" to 0.30". The subject change in tolerances meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The design change made to provide the fabricator with additional flexibility to achieve a flat surface. The fabricator can start with a 1/2" thick plate and does not have to machine it if it meets the flatness tolerance. The minimum allowable thickness is unchanged. The maximum DSC length is controlled separately, so the additional allowed thickness will not affect the cask / DSC interface. The increase in DSC weight due to the potential increase in top shield plug casing plate thickness is extremely negligible compared to the weight of the DSC. The subject tolerance change will not affect the form, fit or function of the top shield plug casing plate, and will not adversely affect the ability of the DSC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the top shield plug casing plate thickness tolerance change does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top Shield Plug Casing Plate Thickness Tolerance Change			72.48 Log No.: SE00026	
D3-DSC-19; 23/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the top shield plug casing plate thickness tolerance change, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this proposed activity. The subject design change allowed the thickness of the top shield plug top casing plate to vary between 0.24" and 0.52" to allow the fabricator flexibility in machining the top shield plug casing plate. The previously allowed range was 0.24" to 0.30". The subject change in tolerances meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). The design change made to provide the fabricator with additional flexibility to achieve a flat surface. The fabricator can start with a 1/2" thick plate and does not have to machine it if it meets the flatness tolerance. The minimum allowable thickness is unchanged. The maximum DSC length is controlled separately, so the additional allowed thickness will not affect the cask / DSC interface. The increase in DSC weight due to the potential increase in top shield plug casing plate thickness is extremely negligible compared to the weight of the DSC. The subject tolerance change will not affect the form, fit or function of the top shield plug casing plate, and will not adversely affect the ability of the DSC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a top shield plug casing plate thickness tolerance change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The top shield plug casing plate thickness tolerance change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Top Shield Plug Casing Plate Thickness Tolerance Change			72.48 Log No.: SE00026	
D3-DSC-19; 23/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the DSC (Dry Shielded Canister) top shield plug casing plate thickness tolerances.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Weld Upgrades **72.48 Log No.: SE00027**

D3-DSC-20; 24/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	R H Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Mark A. Wright M.A. WRIGHT
INDEPENDENT REVIEWER

Date 11/13/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11-19-97

Recommend Approval ✓ Recommend Disapproval ____ Signature: ____ Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ____ Signature: ____ Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ____ No X

Signature: JR Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: .:____

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Weld Upgrades			**72.48 Log No.: SE00027**	
D3-DSC-20; 24/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 2 of 5**

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses weld upgrades to the DSC (Dry Shielded Canister).

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 5.1, 8.1, and 8.2.

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the DSC weld upgrades. The following changes were made to the DSC, which are shown on drawing 84-007-E:

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Weld Upgrades **72.48 Log No.: SE00027**

D3-DSC-20; 24/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

1) A test port was added to the shield plug to demonstrate leak tightness of the shield plug welds. The test port is welded out and vacuum box tested after the shield plug pressure testing is completed.
2) A 5/16" backing fillet was added to the weld between the side casing and top pressure plates.
3) The welds joining the keyway plates were upgraded from 1/4" groove welds to full penetration welds.
4) Added PT requirements to the welds between the casing plate and the lifting lug posts and center post.

The welds were upgraded to allow the shield plug to be pressure tested through the test port to demonstrate leak tightness of the shield plug. The side casing and keyway weldments were upgraded to reduce the likelihood of leakage during final weld-out of the plug. The test port weld is a 3/8" groove weld. Under normal and accident DSC internal pressures, this weld resists the pressure load on the 2.0" diameter lug. The shear stress induced in the weld is minor (less than 1 ksi). The resistance strength of the 3/8" single vee groove weld is 21 ksi, which far exceeds the expected stress in the weld. During the drop accident, this weld resists the 75g acceleration of the 2.0" diameter by 1/2" thick plug. Therefore, the addition of the test port will not adversely affect the integrity of the DSC. The addition of the test port does not affect the fit, form, or function of the DSC. The changes described above are considered upgrades to the DSC design and do not adversely affect the DSC. During the DSC fabrication process final inspection, leakage was observed through a breakdown in the top shield plug welds. The side casing and keyway weldments were upgraded to reduce the likelihood of leakage during the final weld-out of the plug. Since these changes will improve the integrity of the DSC and will not affect any other ISFSI SSC, the proposed activity does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

<u>Consequences of Malfunction:</u>

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

<u>Probability of Accident:</u>

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the proposed activity. One accident scenario described in the ISFSI USAR addresses DSC leakage. The accident would be associated with this activity if the DSC did not meet design requirements and did not maintain integrity. However, the DSC meets it design requirements and will maintain integrity. The DSC design was changed to upgrade the welds on the top and side casing plates. The welds were upgraded to allow the shield plug to be pressure tested through the test port to demonstrate leak tightness of the shield plug. The side casing and keyway weldments were upgraded to reduce the likelihood of leakage during final weld-out of the plug. Under normal and accident DSC internal pressures, this weld resists the pressure load on the 2.0" diameter lug. Therefore, the addition of the test port will not adversely affect the integrity of the DSC. The addition of the test port does not affect the fit, form, or function of the DSC. The changes are considered upgrades to the DSC design and does not adversely affect the DSC. Since the DSC meets it's design requirements and will maintain integrity, the proposed activity will not affect the possibility of occurrence of an accident. Based on the above discussion and a thorough review of all applicable documents, it was concluded that this proposed activity would not increase the probability of occurrence of an accident previously evaluated in the SAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The proposed activity does not affect the resulting dose rate in or around the HSM or DSC. The DSC meets it's design requirements and will maintain it's integrity. Therefore, the ISFSI SSCs will not be adversely affected and will remain intact as designed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of the proposed activity. The subject change does not compromise design integrity, and will not affect the form, fit or function of the DSC. Therefore, this design change has no detrimental impact on equipment important to safety. In regard to the proposed change, no credible scenario can be postulated which could create a malfunction of a different type than any previously evaluated in the SAR. After a thorough review, it was concluded that this activity would not create the possibility of a malfunction of a different type than any previously evaluated in the SAR.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
3/4.2.2	DSC Closure Welds - The proposed activity is a DSC design change which upgraded the welds in the top and side casing plates. It does not affect any other ISFSI SSC. The bases of this technical specification is to ensure that the safety analysis of leak tightness of the DSC is maintained. The safety analysis is based on a weld being leak tight to 10E-4 atm-cc/s. This activity upgrades the welds to ensure that that leak tightness is obtained. Therefore, the proposed activity does not affect this technical specification, and therefore, does not affect the margin of safety associated with this technical specification.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided weld upgrades to the DSC. BGE approved these weld upgrades for construction prior to the issuance of the ISFSI license in November, 1992. The weld upgrades to the DSC do not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Weld Upgrades **72.48 Log No.: SE00027**

D3-DSC-20; 24/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses weld upgrades to the DSC (Dry Shielded Canister).

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Guide Sleeve Inside Dimension Non Conformance **72.48 Log No.: SE00028**

D4-DSC-1; 25/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11.7.97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	R. H. Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A.Wright
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11.19.97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ POSRC CHAIRMAN Date 11-19-97

Approved ✓ Disapproved ______ Signature: ______ PLANT GENERAL MANAGER Date 11-19-97

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: JM Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Guide Sleeve Inside Dimension Non Conformance			72.48 Log No.: SE00028	
D4-DSC-1; 25/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the DSC (Dry Shielded Canister) guide sleeves identified during DSC fabrication. This non conformance applies only to DSC BGE24P-R001.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Guide Sleeve Inside Dimension Non Conformance			72.48 Log No.: SE00028	
D4-DSC-1; 25/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

<u>Complete for 50.59 and 72.48:</u>

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

<u>Probability of Malfunction:</u>

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the guide sleeve inside dimension non conformance. The subject non conformance (Ranor, Inc. NCR No. 9500-3) identifies the DSC guide sleeves having oversize inside dimensions. The allowable dimension is 8.70" +/- 0.03" (see drawing 84-002-E). The maximum recorded deviation is 0.025" over the high tolerance limit. The oversize dimension has no effect on the design as long as the guide sleeves fit in the basket assembly. The fuel assemblies are located in the basket assembly by the spacer disc cutouts and the guide sleeve thickness. Neither of these items are out of tolerance. It must be noted that this non conformance applies only to DSC BGE24P-R001, which was loaded and stored in the HSM in 1996. The minimum possible gap between the inside of the spacer disc cutout and the outside of the guide sleeve is 0.0675" less the finish thickness. This non conformance reduces the possible gap to {0.0675" - (0.025" / 2)} = 0.0675" - 0.0125 = 0.0550". This still leaves enough of a gap for the required minimum 500 micro-inch finish. The subject non conformance meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on the above information and a review of the design drawings, the subject non conformance will not affect the form, fit or function of the DSC, is not detrimental to the structural integrity of the DSC, and will not adversely affect the ability of the DSC to perform it's intended design function. There is no detrimental operational impact associated with this activity. Additionally, the subject justification will not create any component assembly interference, including the guide sleeve and spacer disc interface. Therefore, this activity has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

<u>Consequences of Malfunction:</u>

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

<u>Probability of Accident:</u>

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the guide sleeve inside dimension non conformance does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Guide Sleeve Inside Dimension Non Conformance **72.48 Log No.: SE00028**

D4-DSC-1; 25/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the DSC has not changed as a result of the guide sleeve inside dimension non conformance, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject non conformance identifies the DSC guide sleeves having oversize inside dimensions. It must be noted that this non conformance applies only to DSC BGE24P-R001. The minimum possible gap between the inside of the spacer disc cutout and the outside of the guide sleeve is not reduced as a result of this non conformance. The subject non conformance meets the current design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on the above information and a review of the design drawings, the subject non conformance will not affect the form, fit or function of the DSC, is not detrimental to the structural integrity of the DSC, and will not adversely affect the ability of the DSC to perform it's intended design function. There is no detrimental operational impact associated with this activity. Additionally, the subject justification will not create any component assembly interference, including the guide sleeve and spacer disc interface. Therefore, this activity has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity involved a guide sleeve inside dimension non conformance. BGE approved this non conformance for construction prior to the issuance of the ISFSI license in November, 1992. The guide sleeve inside dimension non conformance does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Guide Sleeve Inside Dimension Non Conformance			72.48 Log No.: SE00028	
D4-DSC-1; 25/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the DSC (Dry Shielded Canister) guide sleeves identified during DSC fabrication. This non conformance applies only to DSC BGE24P-R001.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Bottom Interior Seal Weld Non Conformance **72.48 Log No.: SE00030**

D4-DSC-3; 27/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11·7·97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	R H Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Mark A. Wright M.A.Wright
INDEPENDENT REVIEWER
Date 11/13/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for [✓ GS-DES] GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-132 Date: 11·19·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: Date 11-19-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: Date 11-19-97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Bottom Interior Seal Weld Non Conformance			72.48 Log No.: SE00030	
D4-DSC-3; 27/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the DSC (Dry Shielded Canister) bottom interior seal weld identified during DSC fabrication. This non conformance applies only to DSC Nos. BGE24P-R001, BGE24P-R002, and BGE24P-R003.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Special Note: This proposed activity was presented as a 10 CFR 72.48 safety evaluation to the Plant Operations and Safety Review Committee (POSRC) on April 6, 1992, Meeting No. 92-035. POSRC reviewed and recommended approval of the safety evaluation to the Plant General Manager, who subsequently approved the safety evaluation. Since this safety evaluation was approved prior to the issuance of the ISFSI 10 CFR 72.48 license, the change was incorporated in the first revision of the original SAR. As stated above, this safety evaluation was performed even though the change was incorporated into the ISFSI USAR.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Dry Shielded Canister (DSC) - the DSC is a Type 304 stainless steel cylinder with an internal stainless steel or aluminum coated carbon steel basket assembly that houses 24 fuel assemblies. The DSC is designed to fit securely in the TC and to slide into the HSM from the TC without undue galling. The function of the DSC is to provide physical and radiological protection, and structural support of the spent fuel during loading operations and storage in the HSM. The DSC has been designed for the worst-case postulated accidents, so that retrievability of the fuel from the DSC is assured even following a maximum credible accident.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.2, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Bottom Interior Seal Weld Non Conformance			72.48 Log No.: SE00030	
D4-DSC-3; 27/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of the DSC bottom interior seal weld non conformance. The subject non conformance (Ranor, Inc. NCR No. 9500-6) identifies that the interior 1/4" seal weld at the bottom end of the DSC was not made with at least two passes and at least two levels of PT inspection (see drawing 84-003-E). The subject closure weld was made with a single pass and a single liquid penetrant (PT) inspection was performed on the weld. The PT inspection showed the weld to be satisfactory. It must be noted that this non conformance applies only to DSC Nos. BGE24P-R001, BGE24P-R002, and BGE24P-R003. All other DSC's meet the existing requirement for the weld. The safety function of the DSC is to provide a physical containment barrier to prevent the release of radioactive materials from spent fuel which is stored inside. The double closure welds at each end of the canisters form a part of this physical containment barrier. The structural quality of the double closure seal weld is not affected by the number of passes. The multiple liquid penetrant inspection, which reduces the probability of coincidental pinhole flaws, is compensated by the requirement to leak test the weld. Leak testing the closure weld provides positive assurance of leak tightness. There is no reduction in the structural support or quality of the DSC. The subject non conformance meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on the above information and a review of the design drawings, the subject non conformance is not detrimental to the structural integrity of the DSC and will not adversely affect the ability of the DSC to perform it's intended design function. Leak testing of the closure weld assures leak tightness of the DSC and compensates for the liquid penetrant inspection. Therefore, this activity has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the DSC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the proposed activity. One accident scenario described in the ISFSI USAR addresses DSC leakage. The accident would be associated with this activity if the DSC did not meet design requirements and did not maintain integrity. However, the DSC meets it design requirements and passes it's required acceptance testing. Since the DSC bottom interior seal weld non conformance does not adversely affect the ability of the DSC to perform it's intended design function, the structural integrity of the DSC is not affected, and as such, the probability of occurrence of the DSC leakage accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Bottom Interior Seal Weld Non Conformance			**72.48 Log No.: SE00030**	
D4-DSC-3; 27/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 4 of 5**

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. Since the intended design function of the DSC has not changed as a result of the DSC bottom interior seal weld non conformance, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject non conformance identifies that the interior 1/4" seal weld at the bottom end of the DSC was not made with at least two passes and at least two levels of PT inspection. The subject closure weld was made with a single pass and a single liquid penetrant (PT) inspection was performed on the weld. The PT inspection showed the weld to be satisfactory. It must be noted that this non conformance applies only to DSC Nos. BGE24P-R001, BGE24P-R002, and BGE24P-R003. All other DSC's meet the existing requirement for the weld. The multiple liquid penetrant inspection, which reduces the probability of coincidental pinhole flaws, is compensated by the requirement to leak test the weld. Leak testing the closure weld provides positive assurance of leak tightness. There is no reduction in the structural support or quality of the DSC. The subject non conformance meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on the above information and a review of the design drawings, the subject non conformance is not detrimental to the structural integrity of the DSC and will not adversely affect the ability of the DSC to perform it's intended design function. Leak testing of the closure weld assures leak tightness of the DSC and compensates for the liquid penetrant inspection. Therefore, this activity has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases	Discussion of why the margin of safety is not reduced
3/4.2.2	This technical specification addresses the minimum allowable leak tightness for DSC closure welds. To ensure compliance with this technical specification, the USAR specifies a certain sequence of events including the performance of NDE on the DSC seal welds prior to performance of helium leak testing. This order of operations is consistent with the manufacturer design as detailed in the NUHOMS-24P Topical Report, Section 5.1, Operation Description, which describes the performance of dye penetrant weld examination of the seal weld just after the weld is created. As such, the margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - DSC Bottom Interior Seal Weld Non Conformance			**72.48 Log No.: SE00030**	
D4-DSC-3; 27/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 5 of 5**

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity involved a DSC bottom interior seal weld non conformance. BGE approved this non conformance for construction prior to the issuance of the ISFSI license in November, 1992. The DSC bottom interior seal weld non conformance does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the DSC (Dry Shielded Canister) bottom interior seal weld identified during DSC fabrication. This non conformance applies only to DSC Nos. BGE24P-R001, BGE24P-R002, and BGE24P-R003.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper & Lower Trunnion Design Changes **72.48 Log No.: SE00033**

D3-TC-1; 30/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11.7.97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Jon N Woodfield Jon N.Woodfield
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for GS-DES, ES-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11.24.97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: JR Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper & Lower Trunnion Design Changes			72.48 Log No.: SE00033	
D3-TC-1; 30/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate ISFSI design changes that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses design changes to the TC (Transfer Cask) upper and lower trunnion sleeves.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper & Lower Trunnion Design Changes	**72.48 Log No.: SE00033**
D3-TC-1; 30/129; ES199601368 Supplement 001	**Revision 0000 Page 3 of 5**

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the upper & lower trunnion design changes. The subject activity changed the material for the trunnion sleeves to SA 182 F304N (see drawing 84-021-E). They were 533 Gr B Cl2 or 508 Cl 3A (upper) and 516 Gr 70 or 508 Cl 3A (lower). The outer diameter of the upper trunnion sleeves (see drawing 84-023-E) was changed to 17.0" from 15.15". The subject changes meet the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). The trunnion changes were analyzed in revision 4 of calculation BGE001.0202. The revised trunnion analysis shows that stresses due to the design basis loads remain below allowables. A review of calculation BGE001.0202, Transfer Cask Structural Analysis, revealed that the upper and lower trunnions (with the new material SA 182, F304N) were analyzed for seven load conditions (three handling and four transportation). The total design weight of the transfer cask and DSC is 200k, versus an estimated absolute worst case actual weight of 188.5k. Trunnion stresses were limited to Fy/6 or Fu/10. In addition, all handling cases were increased by 15% for motion loads. This is required per CMAA #70. The revised trunnion design is therefore acceptable from a structural standpoint, and has no operational or radiological impact. Based on this information, the subject design changes will not affect the form, fit or function of the TC trunnions, are not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. These design changes do not affect the lifting or positioning of the transfer cask. Therefore, these design changes have no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the upper & lower trunnion design changes do not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper & Lower Trunnion Design Changes			72.48 Log No.: SE00033	
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NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the upper & lower trunnion design changes, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity changed the material for the trunnion sleeves and the outer diameter of the upper trunnion sleeves was increased. The subject changes meet the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). The trunnion changes were analyzed in revision 4 of calculation BGE001.0202. The revised trunnion analysis shows that stresses due to the design basis loads remain below allowables. The revised trunnion design is therefore acceptable from a structural standpoint, and has no operational or radiological impact. Based on this information, the subject design changes will not affect the form, fit or function of the TC trunnions, are not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. These design changes do not affect the lifting or positioning of the transfer cask. Therefore, these design changes have no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided upper & lower trunnion design changes. BGE approved these design changes for construction prior to the issuance of the ISFSI license in November, 1992. The upper & lower trunnion design changes do not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper & Lower Trunnion Design Changes	72.48 Log No.: SE00033
D3-TC-1; 30/129; ES199601368 Supplement 001 Revision 0000	Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate ISFSI design changes that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses design changes to the TC (Transfer Cask) upper and lower trunnion sleeves.

Reason for Activity: These design changes were fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. These design changes were included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Structural Shell Upper Section Design Change 72.48 Log No.: SE00034

D3-TC-2; 31/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11-7-97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	R. H. Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved [✓ Approved] Disapproved

Signature: Jon N. Woodfield - Jon N. Woodfield Signature: Michael J. Gahan
INDEPENDENT REVIEWER for [✓ GS-DES], GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11/12/97 Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11-24-97

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: J R Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.:________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Structural Shell Upper Section Design Change			72.48 Log No.: SE00034	
D3-TC-2; 31/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) upper trunnion structural shell.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Structural Shell Upper Section Design Change			72.48 Log No.: SE00034	
D3-TC-2; 31/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the upper trunnion structural shell design change. The subject activity involved the replacement of the 2" thick trunnion insert plates with the 2" thick upper shell section (see drawing 84-023-E). The 2" thick portion of the structural shell is equal to, or larger than, the insert plate that it replaces. The penetration stresses calculated in BGE001.0202 are therefore conservative for the 2" thick upper shell and no additional calculations are required. The revised design has no significant radiological or operational impact. A review of calculation BGE001.0202, Transfer Cask Structural Analysis, revealed that the use of a thicker shell in lieu of insert plates will indeed result in a more conservative design. Based on this information, the subject design change will not affect the form, fit or function of the TC upper trunnions, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. The increase in weight of the TC caused by the increased shell thickness is insignificant compared to the weight of the entire TC. This small weight increase would not be detrimental during the lifting or positioning of the TC. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the upper trunnion structural shell design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Structural Shell Upper Section Design Change				72.48 Log No.: SE00034
D3-TC-2; 31/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the upper trunnion structural shell design change, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity involved the replacement of the 2" thick trunnion insert plates with the 2" thick upper shell section. The 2" thick portion of the structural shell is equal to, or larger than, the insert plate that it replaces. The penetration stresses calculated in BGE001.0202 are therefore conservative for the 2" thick upper shell and no additional calculations are required. The revised design has no significant radiological or operational impact. A review of calculation BGE001.0202, Transfer Cask Structural Analysis, revealed that the use of a thicker shell in lieu of insert plates will indeed result in a more conservative design. Based on this information, the subject design change will not affect the form, fit or function of the TC upper trunnions, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. The increase in weight of the TC caused by the increased shell thickness is insignificant compared to the weight of the entire TC. This small weight increase would not be detrimental during the lifting or positioning of the TC. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided an upper trunnion structural shell design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The upper trunnion structural shell design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Structural Shell Upper Section Design Change			72.48 Log No.: SE00034	
D3-TC-2; 31/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) upper trunnion structural shell.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Sleeve Weld Design Change **72.48 Log No.: SE00035**

D3-TC-3; 32/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11-7-97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved [✓ Approved] Disapproved

Signature: Jon N Woodfield - Jon N Woodfield Signature: Michael J. Gahan
INDEPENDENT REVIEWER for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11/12/97 Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11-24-97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: J R Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Sleeve Weld Design Change			72.48 Log No.: SE00035	
D3-TC-3; 32/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) upper trunnion sleeve.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Sleeve Weld Design Change			72.48 Log No.: SE00035	
D3-TC-3; 32/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the upper trunnion sleeve design change. The subject activity deleted the inconel butter layer from the end of the upper trunnion sleeves, and also changed the weldment between the upper trunnion sleeve and the trunnion from a 7/8" "J" weld with a 3/8" fillet to a 1-1/4" "J" weld with a 3/8" fillet (see drawing 84-018-E). The butter layer was no longer needed since the upper trunnion sleeve was changed to stainless steel. The weld size was increased to add strength to the upper trunnion to trunnion sleeve joint. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Inconel butter requirements are not needed for corrosion protection because the trunnion sleeve was changed to stainless steel. The redesigned weld detail is analyzed in calculation BGE001.0202. A review of calculation BGE001.0202, Transfer Cask Structural Analysis, revealed that all actual weld stresses were below the allowables. The welding filler material used was ERNICR-3 or AWS ENICRFE-3. The critical lift analysis yielded the highest actual to allowable stresses in both potential failure planes of 0.88 and 0.66, respectively, where 1.00 is the point that the actuals equal the allowables. The revised design is therefore acceptable from a structural standpoint. The revised design has no operational or radiological impact. Based on this information, the subject design change, deleting of the stainless butter layer and increasing the subject weld size, will not affect the form, fit or function of the TC trunnions, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. This design change does not affect the lifting or positioning of the transfer cask. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the upper trunnion sleeve design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the upper trunnion sleeve design change, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity deleted the inconel butter layer from the end of the upper trunnion sleeves, and also changed the weldment between the upper trunnion sleeve and the trunnion. The butter layer was no longer needed since the upper trunnion sleeve was changed to stainless steel. The weld size was increased to add strength to the upper trunnion to trunnion sleeve joint. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). A review of calculation BGE001.0202, Transfer Cask Structural Analysis, revealed that all actual weld stresses were below the allowables. The welding filler material used was ERNICR-3 or AWS ENICRFE-3. The critical lift analysis yielded the highest actual to allowable stresses in both potential failure planes of 0.88 and 0.66, respectively, where 1.00 is the point that the actuals equal the allowables. The revised design is therefore acceptable from a structural standpoint. The revised design has no operational or radiological impact. Based on this information, the subject design change, deleting of the stainless butter layer and increasing the subject weld size, will not affect the form, fit or function of the TC trunnions, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. This design change does not affect the lifting or positioning of the transfer cask. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided an upper trunnion sleeve design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The upper trunnion sleeve design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Sleeve Weld Design Change				72.48 Log No.: SE00035
D3-TC-3; 32/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

<u>A significant unreviewed environmental impact:</u>

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) upper trunnion sleeve.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Lower Trunnion Sleeve Design Changes			**72.48 Log No.: SE00036**	
D3-TC-4; 33/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 1 of 5**

Based on the attached discussion, does this activity:

<u>Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations</u>

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

<u>Applicable to 10 CFR 72.48 Safety Evaluations</u>

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk JER Department: NED-CEU 42-01-04 Date: 11·7·97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	RH Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved [✓ Approved] Disapproved

Signature: Jon N Woodfield - Jon N. Woodfield Signature: Michael J. Gahan
INDEPENDENT REVIEWER for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11/12/97 Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11·24·97

Recommend Approval ✓ Recommend Disapproval ________ Signature: J. Call Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: P Kat Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: JR Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Lower Trunnion Sleeve Design Changes			72.48 Log No.: SE00036	
D3-TC-4; 33/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate ISFSI design changes that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses design changes to the TC (Transfer Cask) lower trunnion sleeve.

Reason for Activity: These design changes were fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. These design changes were included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Lower Trunnion Sleeve Design Changes			72.48 Log No.: SE00036	
D3-TC-4; 33/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the lower trunnion sleeve design changes. The subject activity deleted the stainless butter layer from the end of the lower trunnion sleeve, and increased the height of the sleeve from 4.25" to 4.5" (see drawing 84-024-E). Since the lower trunnion sleeve was changed to stainless steel, the subject butter layer was no longer needed. The butter layer was used to provide corrosion protection for the carbon steel trunnion sleeve. The height of the lower trunnion sleeve was changed to compensate for the increased thickness of the structural shell upper section. The structural shell upper section thickness was increased by 1/2" to 2", and centerline increase is therefore (1/2") / (2) = 1/4". The subject changes meet the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Revision 4 of calculation BGE001.0202 shows that the stress intensities in the redesigned trunnion are below allowables for each of the design basis loadings. In addition, a review of calculation BGE001.0202, Transfer Cask Structural Analysis, revealed that all actual weld stresses were below the allowables. The revised trunnion design is therefore acceptable from a structural standpoint, and has no operational or radiological impact. Based on this information, the subject design changes will not affect the form, fit or function of the TC trunnions, are not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. These design changes do not affect the lifting or positioning of the transfer cask. Therefore, these design changes have no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the lower trunnion sleeve design changes do not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Lower Trunnion Sleeve Design Changes			72.48 Log No.: SE00036	
D3-TC-4; 33/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the lower trunnion sleeve design changes, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity deleted the stainless butter layer from the end of the lower trunnion sleeve, and increased the height of the sleeve from 4.25" to 4.5". The subject changes meet the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Revision 4 of calculation BGE001.0202 shows that the stress intensities in the redesigned trunnion are below allowables for each of the design basis loadings. In addition, a review of calculation BGE001.0202, Transfer Cask Structural Analysis, revealed that all actual weld stresses were below the allowables. The revised trunnion design is therefore acceptable from a structural standpoint, and has no operational or radiological impact. Based on this information, the subject design changes will not affect the form, fit or function of the TC trunnions, are not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. These design changes do not affect the lifting or positioning of the transfer cask. Therefore, these design changes have no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided lower trunnion sleeve design changes. BGE approved these design changes for construction prior to the issuance of the ISFSI license in November, 1992. The lower trunnion sleeve design changes do not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Lower Trunnion Sleeve Design Changes			72.48 Log No.: SE00036	
D3-TC-4; 33/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate ISFSI design changes that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses design changes to the TC (Transfer Cask) lower trunnion sleeve.

Reason for Activity: These design changes were fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. These design changes were included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Surface Finish Requirements Design Change 72.48 Log No.: SE00037

D3-TC-5; 34/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO Involve an unreviewed safety question (USQ)?
NO Involve a change in the Technical Specifications/License Conditions or Bases?
NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO Involve a Significant Increase in Occupational Dose?
NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11·7·97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved [✓ Approved] Disapproved

Signature: Signature: Michael J. Gahan
INDEPENDENT REVIEWER for [✓ GS-DES], GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11/12/97 Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11·24·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: J R Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Surface Finish Requirements Design Change			72.48 Log No.: SE00037	
D3-TC-5; 34/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) surface finish requirements.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Surface Finish Requirements Design Change			72.48 Log No.: SE00037	
D3-TC-5; 34/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the surface finish requirements design change. The subject activity improved the cask surface finish requirements on all exposed surfaces to 63 micro-inches rms (see drawing 84-021-E). The sole reason for this design change was to improve the TC surface finish to facilitate cask decontamination. This change does not change the structural adequacy of the cask. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on this information, the subject design change will not affect the form, fit or function of the TC structural shell, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. This design change does not affect the lifting or positioning of the transfer cask. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the surface finish requirements design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the surface finish requirements design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Surface Finish Requirements Design Change			72.48 Log No.: SE00037	
D3-TC-5; 34/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity improved the cask surface finish requirements on all exposed surfaces to 63 micro-inches rms. The sole reason for this design change was to improve the TC surface finish to facilitate cask decontamination. This change does not change the structural adequacy of the cask. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on this information, the subject design change will not affect the form, fit or function of the TC structural shell, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. This design change does not affect the lifting or positioning of the transfer cask. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a surface finish requirements design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The surface finish requirements design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Surface Finish Requirements Design Change			72.48 Log No.: SE00037	
D3-TC-5; 34/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) surface finish requirements.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Bottom Cover Plate Design Changes **72.48 Log No.: SE00038**
D3-TC-6; 35/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO Involve an unreviewed safety question (USQ)?
NO Involve a change in the Technical Specifications/License Conditions or Bases?
NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO Involve a Significant Increase in Occupational Dose?
NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Jon N. Woodfield Jon N. Woodfield
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for [✓ GS-DES], GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11·24·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Bottom Cover Plate Design Changes			72.48 Log No.: SE00038	
D3-TC-6; 35/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate ISFSI design changes that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses design changes to the TC (Transfer Cask) bottom cover plate.

Reason for Activity: These design changes were fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. These design changes were included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Bottom Cover Plate Design Changes **72.48 Log No.: SE00038**

D3-TC-6; 35/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 3 of 5**

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the bottom cover plate design changes. The subject activity moved the bottom cover bolt circle out and the seal installation groove in to allow the bottom cover seal to be placed inside the bolt circle (see drawings 84-027-E and 84-030-E). The bolt circle on the temporary shield plug was changed accordingly. The reason for these changes was to reduce the likelihood of leakage through the cask bottom cover. The subject changes meet the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). These changes do not change the structural adequacy of the cask. The bottom cover plate assembly is to be used for transfer cask operations within the Auxiliary Building. The temporary shield plug is to be installed for all cask operations outside of the Auxiliary Building during which spent fuel is present. The design changes are therefore acceptable from a structural standpoint, and have no operational or radiological impact. Based on this information, the subject design changes will not affect the form, fit or function of the TC bottom cover plate, are not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. These design changes do not affect the lifting or positioning of the transfer cask. Therefore, these design changes have no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the bottom cover plate design changes do not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Bottom Cover Plate Design Changes **72.48 Log No.: SE00038**

D3-TC-6; 35/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 4 of 5**

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the bottom cover plate design changes, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity moved the bottom cover bolt circle out and the seal installation groove in to allow the bottom cover seal to be placed inside the bolt circle. The bolt circle on the temporary shield plug was change accordingly. The reason for these changes was to reduce the likelihood of leakage through the cask bottom cover. The subject changes meet the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). These changes do not change the structural adequacy of the cask. The design changes have no operational or radiological impact. Based on this information, the subject design changes will not affect the form, fit or function of the TC bottom cover plate, are not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. These design changes do not affect the lifting or positioning of the transfer cask. Therefore, these design changes have no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided bottom cover plate design changes. BGE approved these design changes for construction prior to the issuance of the ISFSI license in November, 1992. The bottom cover plate design changes do not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Bottom Cover Plate Design Changes 72.48 Log No.: SE00038

D3-TC-6; 35/129; ES199601368 Supplement 001 Revision 0000 Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate ISFSI design changes that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses design changes to the TC (Transfer Cask) bottom cover plate.

Reason for Activity: These design changes were fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. These design changes were included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Attachment Design Change **72.48 Log No.: SE00039**

D3-TC-7; 36/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11.7.97

YES	Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Jon N Woodfield - Jon N. Woodfield
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11.24.97

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: ________ Date: 1/30/98
OSSRC/SES CHAIRMAN

If yes, OSSRC Meeting No.: ________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Attachment Design Change			72.48 Log No.: SE00039	
D3-TC-7; 36/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) upper trunnion covers.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Attachment Design Change			72.48 Log No.: SE00039	
D3-TC-7; 36/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the upper trunnion attachment design change. The subject activity removed the tapped holes for the upper trunnion covers and added a weld between the trunnion and cover (see drawing 84-029-E). The reason for this design change was to eliminate the trapping of crud between the cover plate and trunnion, thus easing cask decontamination. The method of attachment for the upper trunnion covers was changed from bolting to welding (5/16" all-around fillet weld). The gap between the cover and the trunnion was thus removed, easing the decontamination of the cask. The weld material provides equivalent strength to the bolts that were replaced. This change therefore, has no negative impact on the structural adequacy of the cask. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on this information, the subject design change will not affect the form, fit or function of the TC upper trunnions, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. This design change does not affect the lifting or positioning of the transfer cask. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the upper trunnion attachment design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ISFSI - Upper Trunnion Attachment Design Change **72.48 Log No.: SE00039**

D3-TC-7; 36/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the upper trunnion attachment design change, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity removed the tapped holes for the upper trunnion covers and added a weld between the trunnion and cover. The reason for this design change was to eliminate the trapping of crud between the cover plate and trunnion, thus easing cask decontamination. The method of attachment for the upper trunnion covers was changed from bolting to welding (5/16" all-around fillet weld). The gap between the cover and the trunnion was thus removed, easing the decontamination of the cask. The weld material provides equivalent strength to the bolts that were replaced. This change therefore, has no negative impact on the structural adequacy of the cask. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on this information, the subject design change will not affect the form, fit or function of the TC upper trunnions, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. This design change does not affect the lifting or positioning of the transfer cask. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a upper trunnion attachment design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The upper trunnion attachment design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Trunnion Attachment Design Change **72.48 Log No.: SE00039**

D3-TC-7; 36/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 5 of 5**

NO Will the proposed activity involve a significant unreviewed environmental impact?

<u>A significant unreviewed environmental impact:</u>

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) upper trunnion covers.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Flange Relief Holes Threading **72.48 Log No.: SE00040**

D3-TC-8; 37/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye	Resp. Indv.: C. J. Dobry	Resp. Indv.: R. H. Beall
Work Group: Licensing	Work Group: PES	Work Group: NFM
11/12/97	11/10/97	11/11/97
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Jon N. Woodfield - Jon N. Woodfield
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for ~~GS-DES~~, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11-24-97

Recommend Approval ✓ Recommend Disapproval ____ Signature: ____ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ____ Signature: ____ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ____ No X

Signature: JR Lemon Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ____

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Flange Relief Holes Threading			72.48 Log No.: SE00040	
D3-TC-8; 37/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) top flange.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Flange Relief Holes Threading			72.48 Log No.: SE00040	
D3-TC-8; 37/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the top flange relief holes threading design change. The subject activity added threads to the relief holes in the TC top flange to allow them to be plugged when the cask is immersed in the fuel pool (see drawing 84-022-E). This helps ease the decontamination of the top cover bolt holes before installation of the cover. Water relief holes are tapped 3/8"-16 UNC-2B x .50" deep, and are provided at each pin and bolt hole, drilled horizontally to meet bottom of the vertical holes. Based on this information, the subject design change will not affect the form, fit or function of the TC top flange or the flange to top cover plate joint interface, is not detrimental to the structural integrity of the TC and will not adversely affect the ability of the TC to perform it's intended design function. This design change enhanced TC design, in that, it reduces the potential for the relief holes to become contaminated. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the top flange relief holes threading design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the top flange relief holes threading design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Flange Relief Holes Threading			72.48 Log No.: SE00040	
D3-TC-8; 37/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity added threads to the relief holes in the TC top flange to allow them to be plugged when the cask is immersed in the fuel pool. This helps ease the decontamination of the top cover bolt holes before installation of the cover. Based on this information, the subject design change will not affect the form, fit or function of the TC top flange or the flange to top cover plate joint interface, is not detrimental to the structural integrity of the TC and will not adversely affect the ability of the TC to perform it's intended design function. This design change enhanced TC design, in that, it reduces the potential for the relief holes to become contaminated. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a top flange relief holes threading design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The top flange relief holes threading design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Flange Relief Holes Threading			72.48 Log No.: SE00040	
D3-TC-8; 37/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) top flange.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shell Weld Process Design Change **72.48 Log No.: SE00041**

D3-TC-9; 38/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye	Resp. Indv.: C. J. Dobry	Resp. Indv.: R. H. Beall
Work Group: Licensing	Work Group: PES	Work Group: NFM
11/12/97	11/10/97	R H Beall 11/11/97
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Jon N. Woodfield Jon N. Woodfield
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for [✓ GS-DES], GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11-24-97

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: J R Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shell Weld Process Design Change			72.48 Log No.: SE00041	
D3-TC-9; 38/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) structural shell weld process.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shell Weld Process Design Change			72.48 Log No.: SE00041	
D3-TC-9; 38/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the shell weld process design change. The subject activity allowed the use of automatic submerged arc weld process for weldments between structural shell and forgings, with proper protection of the heat affected zone. The other allowed welding methods were gas tungsten arc and gas metal arc. The reason for this change is to facilitate fabrication of the TC shell. Welds made by the submerged-arc process are found to have uniformly high quality, good ductility, high density, high impact strength, and good corrosion resistance. Mechanical properties of the weld are consistently as good as the base metal. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on this information, the subject design change will not affect the form, fit or function of the TC structural shell, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. This design change does not affect the design properties of the cask or the weld joints. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the shell weld process design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the shell weld process design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity allowed the use of automatic submerged arc weld process for weldments between structural shell and forgings, with proper protection of the heat affected zone. The reason for this change is to facilitate fabrication of the TC shell. Welds made by the submerged-arc process are found to have uniformly high quality, good ductility, high density, high impact strength, and good corrosion resistance. Mechanical properties of the weld are consistently as good as the base metal. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on this information, the subject design change will not affect the form, fit or function of the TC structural shell, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. This design change does not affect the design properties of the cask or the weld joints. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a shell weld process design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The shell weld process design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shell Weld Process Design Change **72.48 Log No.: SE00041**

D3-TC-9; 38/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 5 of 5**

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) structural shell weld process.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Flange Location Hole Depth Design Change 72.48 Log No.: SE00042

D3-TC-10; 39/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11-7-97
PRINTED NAME AND SIGNATURE

YES	Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Jon N. Woodfield Jon N. Woodfield
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for CS-DES CS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11-24-97

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: ________ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ________________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Flange Location Hole Depth Design Change			72.48 Log No.: SE00042	
D3-TC-10; 39/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) top flange location pin hole.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Flange Location Hole Depth Design Change 72.48 Log No.: SE00042

D3-TC-10; 39/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the top flange location hole depth design change. The subject activity changed the length of the location pin hole at the 185 degree azimuth from 1.75" to 2.75" (see drawing 84-022-E). This depth is now consistent with the depth of the location pin hole at the 5 degree azimuth. The reason for this change is to assure adequate depth of the location pin, and to maintain consistency with the depth of the other location pin hole, since the hole at 5 degree azimuth was already designed for 2.75" with a water relief hole at the end of the pin hole. This change does not change the structural adequacy of the cask. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on this information, the subject design change will not affect the form, fit or function of the TC top flange, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. This design change does not affect the lifting or positioning of the transfer cask. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the top flange location hole depth design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the top flange location hole depth design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Flange Location Hole Depth Design Change			**72.48 Log No.: SE00042**	
D3-TC-10; 39/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 4 of 5**

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity changed the length of the location pin hole at the 185 degree azimuth to 2.75". This depth is now consistent with the depth of the location pin hole at the 5 degree azimuth. The reason for this change is to assure adequate depth of the location pin, and to maintain consistency with the depth of the other location pin hole, since the hole at 5 degree azimuth was already designed for 2.75" with a water relief hole at the end of the pin hole. This change does not change the structural adequacy of the cask. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on this information, the subject design change will not affect the form, fit or function of the TC top flange, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. This design change does not affect the lifting or positioning of the transfer cask. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a top flange location hole depth design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The top flange location hole depth design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Flange Location Hole Depth Design Change				**72.48 Log No.: SE00042**
D3-TC-10; 39/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 5 of 5**

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) top flange location pin hole.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of TC Lead Casting Full Surface Requirement **72.48 Log No.: SE00043**

D3-TC-12; 40/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11·7·97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	R H Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Jon N. Woodfield Jon N. Woodfield
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for [✓ GS-DES], GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11·24·97

Recommend Approval ✓ Recommend Disapproval _______ Signature: _______ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved _______ Signature: _______ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes _______ No X

Signature: JR Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: __________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of TC Lead Casting Full Surface Requirement			72.48 Log No.: SE00043	
D3-TC-12; 40/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) lead shielding inspection requirement.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of TC Lead Casting Full Surface Requirement			**72.48 Log No.: SE00043**	
D3-TC-12; 40/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 3 of 5**

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the deletion of the lead casting full surface requirement design change. The subject design change deleted the requirement that the lead casting have full surface contact with the structural shell to facilitate fabrication and pouring of the TC lead shielding. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Full surface contact between the lead casting and the cask shell is neither necessary nor detectable, since any gap between the lead and the shell would not form a streaming path due to the geometry of the cask. The gamma scan required by the fabrication specification ensures that full shielding thickness is obtained. This change therefore does not affect the design or operation of the cask, and does not impact any safety or licensing criteria. Based on the above information, the subject design change will not have a detrimental impact on the integrity or shielding capability of the TC. The subject design change will not affect the form, fit or function of the lead shielding and will not adversely affect the ability of the TC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the deletion of the lead casting full surface requirement design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of TC Lead Casting Full Surface Requirement **72.48 Log No.: SE00043**

D3-TC-12; 40/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 4 of 5**

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the deletion of the lead casting full surface requirement design change, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject design change deleted the requirement that the lead casting have full surface contact with the structural shell to facilitate fabrication and pouring of the TC lead shielding. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Full surface contact between the lead casting and the cask shell is neither necessary nor detectable, since any gap between the lead and the shell would not form a streaming path due to the geometry of the cask. The gamma scan required by the fabrication specification ensures that full shielding thickness is obtained. This change therefore does not affect the design or operation of the cask, and does not impact any safety or licensing criteria. Based on the above information, the subject design change will not have a detrimental impact on the integrity or shielding capability of the TC. The subject design change will not affect the form, fit or function of the lead shielding and will not adversely affect the ability of the TC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity involved the deletion of the lead casting full surface requirement. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The deletion of the lead casting full surface requirement design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1, since the gamma scan required by the fabrication specification ensured that full shielding thickness was obtained.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Deletion of TC Lead Casting Full Surface Requirement			**72.48 Log No.: SE00043**	
D3-TC-12; 40/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 5 of 5**

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) lead shielding inspection requirement.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Neutron Shield Panel Tolerance Design Change **72.48 Log No.: SE00044**

D3-TC-13; 41/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11·7·97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Jon N. Woodfield
Signature: Jon N. Woodfield
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for [✓ GS-DES] GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11·24·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Neutron Shield Panel Tolerance Design Change			72.48 Log No.: SE00044	
D3-TC-13; 41/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) upper neutron shield panel support ring.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Neutron Shield Panel Tolerance Design Change			72.48 Log No.: SE00044	
D3-TC-13; 41/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the upper neutron shield panel support ring tolerance design change. The subject activity loosened the tolerance on the placement of the upper neutron shield panel support ring from +/- 0.06" to +/- 0.12" (see drawings 84-018-E and 84-025-E). The purpose of the old tolerance was to prevent an interference of the weld between the supporting ring and the structural shell with the access port cover. This purpose is now achieved by adding a note to the weldment requiring the weld to be a 5/16" seal weld only where adjacent to access hole cover. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on this information, the subject design change will not affect the form, fit or function of the TC shell, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC/upper neutron shield panel support ring from performing their intended design functions. There is no detrimental operational impact associated with this design change. Additionally, the revised tolerance dimensions will not create any component assembly interference. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the upper neutron shield panel support ring tolerance design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Neutron Shield Panel Tolerance Design Change			72.48 Log No.: SE00044	
D3-TC-13; 41/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the upper neutron shield panel support ring tolerance design change, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity loosened the tolerance on the placement of the upper neutron shield panel support ring from +/- 0.06" to +/- 0.12". The purpose of the old tolerance was to prevent an interference of the weld between the supporting ring and the structural shell with the access port cover. This purpose is now achieved by adding a note to the weldment requiring the weld to be a 5/16" seal weld only where adjacent to access hole cover. The subject change meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). Based on this information, the subject design change will not affect the form, fit or function of the TC shell, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC/upper neutron shield panel support ring from performing their intended design functions. There is no detrimental operational impact associated with this design change. Additionally, the revised tolerance dimensions will not create any component assembly interference. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a upper neutron shield panel support ring tolerance design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The upper neutron shield panel support ring tolerance design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Upper Neutron Shield Panel Tolerance Design Change **72.48 Log No.: SE00044**

D3-TC-13; 41/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 5 of 5**

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) upper neutron shield panel support ring.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Cover Plate Material Design Change **72.48 Log No.: SE00045**

D3-TC-14; 42/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11·7·97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 **SIGNATURE / DATE**	11/10/97 **SIGNATURE / DATE**	11/11/97 **SIGNATURE / DATE**

[✓ Approved] Disapproved

Signature: Jon N Woodfield Jon N. Woodfield
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for **[✓ GS-DHS], GS-TES, or PE-PDSU**
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11·24·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: JR Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Cover Plate Material Design Change	72.48 Log No.: SE00045
D3-TC-14; 42/129; ES199601368 Supplement 001 Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) top cover plate.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Cover Plate Material Design Change			72.48 Log No.: SE00045	
D3-TC-14; 42/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the top cover plate material design change. The subject activity changed the material for the TC top cover plate from carbon steel ASTM A516 Gr 70 with stainless steel ASTM A240 Type 304 to reduce the probability of corrosion of the top cover plate and improve the overall operability of the cask (see drawing 84-027-E). The structural impact of the change is negligible and justified in calculation BGE001.0202 revision 4. The change in material results in a negligible effect on the dead weight (0.286 vs. 0.283 lbs./cu.ft.). For the static analysis performed, the reduction in Modulus of Elasticity E (26.5E6 vs. 27.7 E6) and the increased coefficient of thermal expansion (9.80 E-6 vs. 7.60 E-6) resulted in a reduction of the calculated stresses. Based on this information, the subject design change will not affect the form, fit or function of the TC top cover plate, is not detrimental to the structural integrity of the TC or the top plate joint interface, and will not adversely affect the ability of the TC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the top cover plate material design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the top cover plate material design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Cover Plate Material Design Change 72.48 Log No.: SE00045

D3-TC-14; 42/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity changed the material for the TC top cover plate from carbon steel ASTM A516 Gr 70 with stainless steel ASTM A240 Type 304 to reduce the probability of corrosion of the top cover plate and improve the overall operability of the cask. The structural impact of the change is negligible and justified in calculation BGE001.0202 revision 4. Based on this information, the subject design change will not affect the form, fit or function of the TC top cover plate, is not detrimental to the structural integrity of the TC or the top plate joint interface, and will not adversely affect the ability of the TC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a top cover plate material design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The top cover plate material design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Top Cover Plate Material Design Change			72.48 Log No.: SE00045	
D3-TC-14; 42/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) top cover plate.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Alignment Mounting Holes **72.48 Log No.: SE00046**

D3-TC-15; 43/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11-7-97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Jon N. Woodfield Jon N. Woodfield
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for ~~GS-DES~~, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11-24-97

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: ________ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Alignment Mounting Holes			**72.48 Log No.: SE00046**	
D3-TC-15; 43/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 2 of 5**

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask).

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Alignment Mounting Holes **72.48 Log No.: SE00046**

D3-TC-15; 43/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the alignment mounting holes design change. The subject activity added mounting holes to provide locations for mounting the cask alignment targets (see drawings 84-027-E and 84-029-E). The structural integrity of the cask is not affected. Based on this information, this activity will not affect the form, fit or function of the TC, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. There is no detrimental operational impact associated with this design change. Additionally, this design change will not create any component assembly interference. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the alignment mounting holes design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the alignment mounting holes design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Alignment Mounting Holes			72.48 Log No.: SE00046	
D3-TC-15; 43/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity added mounting holes to provide locations for mounting the cask alignment targets. The structural integrity of the cask is not affected. Based on this information, this activity will not affect the form, fit or function of the TC, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. There is no detrimental operational impact associated with this design change. Additionally, this design change will not create any component assembly interference. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a alignment mounting holes design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The alignment mounting holes design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Alignment Mounting Holes			**72.48 Log No.: SE00046**	
D3-TC-15; 43/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 5 of 5**

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask).

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Top Plate Weld Surface Finish Requirements Clarification 72.48 Log No.: SE00047

D3-TC-16; 44/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO Involve an unreviewed safety question (USQ)?
NO Involve a change in the Technical Specifications/License Conditions or Bases?
NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO Involve a Significant Increase in Occupational Dose?
NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11·7·97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Jon N Woodfield Jon N. Woodfield
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11·24·97

Recommend Approval ✓ Recommend Disapproval ____ Signature: ____ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ____ Signature: ____ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ____ No X

Signature: Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ____

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Top Plate Weld Surface Finish Requirements Clarification			72.48 Log No.: SE00047	
D3-TC-16; 44/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) surface finish requirements for the top cover plate welds.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Top Plate Weld Surface Finish Requirements Clarification 72.48 Log No.: SE00047

D3-TC-16; 44/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the surface finish requirements design change. The subject activity clarified the surface finish requirements of the TC top cover welds for fabrication purposes. Essentially, all exposed external cask, interior cavity, and top and bottom cover plate assembly surfaces shall be finished to 63 (micro) inch RMS or better (see drawing 84-028-E). Plate surfaces which will not be exposed to pool water shall have an ASTM A480 No. 1 or 250 (micro) inch RMS finish. Top cover plate assembly welds are not exposed to the spent fuel pool and need not meet surface finish requirements. These welds shall be ground to permit NDE as required. The subject clarification of the TC top cover plate weld surface finish meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). These welds are not exposed to the pool and therefore need only be ground as required for NDE. This change does not affect the cask design basis. Based on this information, the subject surface finish requirement clarification will not affect the form, fit or function of the TC or the TC top cover plate, is not detrimental to the structural integrity of the TC and will not adversely affect the ability of the TC to perform it's intended design function. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the surface finish requirements design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Top Plate Weld Surface Finish Requirements Clarification 72.48 Log No.: SE00047

D3-TC-16; 44/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the surface finish requirements design change, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity clarified the surface finish requirements of the TC top cover welds for fabrication purposes. The subject clarification of the TC top cover plate weld surface finish meets the original design requirements as established by Pacific Nuclear Fuel Services (PNFS). These welds are not exposed to the pool and therefore need only be ground as required for NDE. This change does not affect the cask design basis. Based on this information, the subject surface finish requirement clarification will not affect the form, fit or function of the TC or the TC top cover plate, is not detrimental to the structural integrity of the TC and will not adversely affect the ability of the TC to perform it's intended design function. There is no detrimental operational impact associated with this design change. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a surface finish requirements design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The surface finish requirements design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Top Plate Weld Surface Finish Requirements Clarification			72.48 Log No.: SE00047	
D3-TC-16; 44/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) surface finish requirements for the top cover plate welds.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shield Plug Material Design Change **72.48 Log No.: SE00048**

D3-TC-17; 45/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO Involve an unreviewed safety question (USQ)?
NO Involve a change in the Technical Specifications/License Conditions or Bases?
NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO Involve a Significant Increase in Occupational Dose?
NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11-7-97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 **SIGNATURE / DATE**	11/10/97 **SIGNATURE / DATE**	11/11/97 **SIGNATURE / DATE**

[✓ Approved] Disapproved

Signature: Jon N. Woodfield Jon N. Woodfield
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for **GS-DES, GS-TES, or PE-PDSU**
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11-24-97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: ______ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shield Plug Material Design Change			72.48 Log No.: SE00048	
D3-TC-17; 45/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) shield plug plate material.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shield Plug Material Design Change **72.48 Log No.: SE00048**

D3-TC-17; 45/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the shield plug plate material design change. The subject activity allowed the use of ASTM A36 or A516 Gr 70 in place of ASTM A283 Grade C plate in the shield plug assembly to provide flexibility in shield plug fabrication (see drawing 84-030-E). The alternate materials are acceptable since they have equal or better allowable stresses, and since the assembly plates are essentially unstressed in this application. This is an acceptable practice to use materials of comparable properties. All three are carbon steels. A36 is a primary structural steel (Fy = 36 ksi), A516 is a pressure vessel steel (Fy = 38 ksi), and A283 is a low tensile strength carbon steel (Fy = 30 ksi). The temporary shield plug assembly is non-safety related. Based on this information, changing the subject temporary shield plug assembly material will not affect the form, fit or function of the TC temporary shield plug, is not detrimental to the structural integrity of the TC or the shield plug, and will not adversely affect the ability of the TC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the shield plug plate material design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the shield plug plate material design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shield Plug Material Design Change			72.48 Log No.: SE00048	
D3-TC-17; 45/129;	ES199601368	Supplement 001	Revision 0000	Page 4 of 5

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity allowed the use of ASTM A36 or A516 Gr 70 in place of ASTM A283 Grade C plate in the shield plug assembly to provide flexibility in shield plug fabrication. The alternate materials are acceptable since they have equal or better allowable stresses, and since the assembly plates are essentially unstressed in this application. Based on this information, changing the subject temporary shield plug assembly material will not affect the form, fit or function of the TC temporary shield plug, is not detrimental to the structural integrity of the TC or the shield plug, and will not adversely affect the ability of the TC to perform it's intended design function. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a shield plug plate material design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The shield plug plate material design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shield Plug Material Design Change			72.48 Log No.: SE00048	
D3-TC-17; 45/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) shield plug plate material.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shield Plug Width Tolerance Design Change **72.48 Log No.: SE00049**

D3-TC-18; 46/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11-7-97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Jon N. Woodfield / Jon N. Woodfield
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-134 Date: 11·24·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-24-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 11/25/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: M. Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shield Plug Width Tolerance Design Change 72.48 Log No.: SE00049

D3-TC-18; 46/129; ES199601368 Supplement 001 Revision 0000 Page 2 of 5

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) shield plug.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shield Plug Width Tolerance Design Change 72.48 Log No.: SE00049

D3-TC-18; 46/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the shield plug tolerance design change. The subject activity relaxed the tolerance requirements on the width of the shield plug assembly inner plug (was +/- .03", now +/- .06"), inner plug support bracket (was 5.00" +/- .03" now "to be free sliding"), and inner diameter of outer plug (was +/- .06", now +/- .12") (see drawing 84-030-E). The reason for this design change was to provide flexibility in shield plug fabrication. The new tolerances are consistent with the functional requirements of the components. The prime consideration is that the components fit together without binding. The shield plug assembly is non-safety related. Based on this information, changing the subject temporary shield plug tolerances will not affect the form, fit or function of the TC temporary shield plug, is not detrimental to the structural integrity of the TC or the shield plug, and will not adversely affect the ability of the TC to perform it's intended design function. Additionally, the revised clearance dimensions will not create any component assembly interference. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the shield plug tolerance design change does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the shield plug tolerance design change, there will be no increase in the accident dose consequences already described in the USAR.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shield Plug Width Tolerance Design Change			**72.48 Log No.: SE00049**	
D3-TC-18; 46/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 4 of 5**

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject activity relaxed the tolerance requirements on the width of the shield plug assembly inner plug, inner plug support bracket, and inner diameter of outer plug. The reason for this design change was to provide flexibility in shield plug fabrication. The new tolerances are consistent with the functional requirements of the components. The prime consideration is that the components fit together without binding. The shield plug assembly is non-safety related. Based on this information, changing the subject temporary shield plug tolerances will not affect the form, fit or function of the TC temporary shield plug, is not detrimental to the structural integrity of the TC or the shield plug, and will not adversely affect the ability of the TC to perform it's intended design function. Additionally, the revised clearance dimensions will not create any component assembly interference. Therefore, this design change has no detrimental impact on equipment important to safety.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity provided a shield plug tolerance design change. BGE approved this design change for construction prior to the issuance of the ISFSI license in November, 1992. The shield plug tolerance design change does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1. This design change did not reduce the lead shielding thickness nor did it alter the shielding capability of the TC. Therefore, this subject design change will not decrease the shielding requirements/ability of the TC.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Shield Plug Width Tolerance Design Change	72.48 Log No.: SE00049
D3-TC-18; 46/129; ES199601368 Supplement 001 Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI design change that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a design change to the TC (Transfer Cask) shield plug.

Reason for Activity: This design change was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This design change was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Minimum Shell Thickness Non Conformance 72.48 Log No.: SE00050

D4-TC-1; 47/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11-7-97
PRINTED NAME AND SIGNATURE

YES	Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97	11/10/97	11/11/97
SIGNATURE / DATE	SIGNATURE / DATE	SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Ton N. Woodfield Ton N. Woodfield
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J. Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meening No.: 97-135 Date: 11·26·97

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 11-26-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 12/1/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: ________ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Minimum Shell Thickness Non Conformance			72.48 Log No.: SE00050	
D4-TC-1; 47/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the TC (Transfer Cask) shell identified during TC fabrication.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Minimum Shell Thickness Non Conformance 72.48 Log No.: SE00050

D4-TC-1; 47/129; ES199601368 Supplement 001 Revision 0000 Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the minimum shell thickness non conformance. The subject non conformance (Sulzer Bingham NCR No. 108826) identifies the TC structural shell as-built plate average thickness to be 1.459" at one of thirty-four measured areas. The minimum allowable thickness of 1.490" was not met. Calculation BGE001.0202, revision 4, shows that the maximum calculated stress versus allowable for the transfer cask structural shell occurs for the Level A Cases 1 through 5 load combinations. The corresponding maximum calculated stress is 55.8 ksi with an allowable of 56.1 ksi. The SA 240 Type 304 plate material for the structural shell has a yield strength of 42.5 ksi and a tensile strength of 89.0 ksi at room temperature, as determined by a CMTR (Certified Material Test Report). This compares with the ASME code minimum values for yield strength of 30 ksi and a tensile strength of 75 ksi used for design.

The Code allowable stress intensity for the plate materials is proportional to the material strength properties. Conservatively assuming that the increased stress in the reduced plate section is resisted entirely by bending, and that the bending stress is inversely proportional to the square of the plate thickness, the minimum acceptable material thickness is determined as follows:

$\{(\text{tmin}) / (1.50)\}^2 = \{S_{design} / S_{actual}\}$

$t_{min} >/= 1.50 \{30.0 / 42.5\}^{1/2}$

$t_{min} >/= 1.26$ inches

Substituting based on tensile strength:

$t_{min} >/= 1.50\{75 / 89\}^{1/2}$

$t_{min} >/= 1.38$ inches

Since the actual thickness of the structural shell exceeds the minimum required thickness, the structural shell is acceptable as is. The reduced shell thickness has a negligible affect on the thermal and shielding calculations. A review of the calculation showed that the design was based on a shell thickness of 1.50", not the minimum required 1.490". However, there are several cases throughout the calculations that the expected loads were conservatively increased (a common practice in design). For example, the total design weight of the transfer cask and DSC is 200k, versus an estimated absolute worst case actual weight of 188.5k. In addition, the transfer cask analytical models were developed and analyzed using a carbon steel SA 516 Gr. 70 shell. The fabricator elected to use a stainless steel SA 240 Type 304 shell. This resulted in lower calculated stresses. Also, the minimum average value of 1.459" was only found in one of thirty-four measured areas. All other areas measured at least 1.472". Based on the above information, the subject non conformance will not affect the form, fit or function of the TC shell, is not detrimental to the structural integrity of the TC and will not adversely affect the ability of the TC to perform it's intended design function. Therefore, this activity has no detrimental impact on equipment important to safety.

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Minimum Shell Thickness Non Conformance 72.48 Log No.: SE00050

D4-TC-1; 47/129; ES199601368 Supplement 001 Revision 0000 Page 4 of 5

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the minimum shell thickness non conformance does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the minimum shell thickness non conformance, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject non conformance identifies the TC structural shell as-built plate thickness to be 1.459". The minimum allowable thickness of 1.490" was not met. The minimum acceptable material thickness was then calculated to be 1.38", which exceeds the minimum required thickness, thus the structural shell is acceptable as is. Based on the above information, the subject non-conformance will not affect the form, fit or function of the TC shell, is not detrimental to the structural integrity of the TC and will not adversely affect the ability of the TC to perform it's intended design function. Therefore, this activity has no detrimental impact on equipment important to safety, and does not create the possibility of a new malfunction.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Minimum Shell Thickness Non Conformance **72.48 Log No.: SE00050**

D4-TC-1; 47/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 5 of 5**

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity involved a minimum shell thickness non conformance. BGE approved this non conformance for construction prior to the issuance of the ISFSI license in November, 1992. The minimum shell thickness non conformance does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the TC (Transfer Cask) shell identified during TC fabrication.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Shell Maximum Lead Thickness Non Conformance **72.48 Log No.: SE00051**
D4-TC-2; 48/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk Department: NED-CEU 42-01-04 Date: 11-7-97
PRINTED NAME AND SIGNATURE

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Jon N. Woodfield Jon N. Woodfield
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for [✓ GS-DES] GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-135 Date: 11·26·97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-26-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 12/1/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: JR Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Shell Maximum Lead Thickness Non Conformance **72.48 Log No.: SE00051**

D4-TC-2; 48/129; ES199601368 Supplement 001 Revision 0000 Page 2 of 5

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the TC (Transfer Cask) shell identified during TC fabrication.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Shell Maximum Lead Thickness Non Conformance			72.48 Log No.: SE00051	
D4-TC-2; 48/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the maximum lead thickness non conformance. The subject non conformance (Sulzer Bingham NCR No. 108831) identifies the TC lead cavity exceeding the maximum allowable thickness. The maximum measured thickness is 4.138" while the maximum allowable is 4.12". A slight increase in the transfer cask weight will result from the increased lead cavity thickness. Calculation BGE001.0202, Revision 4, is based on a total weight of 200 kips. The actual weight of the transfer cask plus the DSC (dry) is 180 kips. The 20 kip weight margin is more than adequate to accommodate the increased lead thickness. Also, the average lead cavity thickness is within the nominal design thickness. The transfer cask is therefore structurally adequate. The thickness increase has a negligible effect on the transfer cask thermal calculations and a positive effect on the shielding calculations. The estimated absolute worst case actual weight is 188.5k, which occurs during the critical vertical handling condition at the spent fuel pool. The 180k referenced above occurs with the cask loaded with the DSC and fuel assemblies during transfer. Still, the design is more than adequate even with the increased lead cavity thickness. Based on the above information, the subject non conformance will not affect the form, fit or function of the TC shell, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. Therefore, this activity has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the maximum lead thickness non conformance does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Shell Maximum Lead Thickness Non Conformance			**72.48 Log No.: SE00051**	
D4-TC-2; 48/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 4 of 5**

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the maximum lead thickness non conformance, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject non conformance identifies the TC lead cavity exceeding the maximum allowable thickness. The maximum measured thickness is 4.138" while the maximum allowable is 4.12". The thickness increase has a negligible effect on the transfer cask structural and thermal calculations, and a positive effect on the shielding calculations. Based on the above information, the subject non conformance will not affect the form, fit or function of the TC shell, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform it's intended design function. Therefore, this activity has no detrimental impact on equipment important to safety, and does not create the possibility of a new malfunction.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity involved a maximum lead thickness non conformance. BGE approved this non conformance for construction prior to the issuance of the ISFSI license in November, 1992. The maximum lead thickness non conformance does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Shell Maximum Lead Thickness Non Conformance			**72.48 Log No.: SE00051**	
D4-TC-2; 48/129;	**ES199601368**	**Supplement 001**	**Revision 0000**	**Page 5 of 5**

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the TC (Transfer Cask) shell identified during TC fabrication.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Top Flange Maximum Bore Diameter Non Conformance 72.48 Log No.: SE00052

D4-TC-3; 49/129; ES199601368 Supplement 001 Revision 0000 Page 1 of 5

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO Involve an unreviewed safety question (USQ)?
NO Involve a change in the Technical Specifications/License Conditions or Bases?
NO Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO Involve a Significant Increase in Occupational Dose?
NO Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
Getachew Tesfaye 11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	RH Beall 11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved
Signature: Jon N. Woodfield Jon N. Woodfield
INDEPENDENT REVIEWER
Date 11/12/97

[✓ Approved] Disapproved
Signature: Michael J. Gahan
for ~~GS-DES~~ GS-TES, or PE-PDSU
MICHAEL J. GAHAN III
Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-135 Date: 11-26-97

Recommend Approval ✓ Recommend Disapproval ______ Signature: ______ Date 11-26-97
POSRC CHAIRMAN

Approved ✓ Disapproved ______ Signature: ______ Date 12/1/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ______ No X

Signature: JR Lemons Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ______

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Top Flange Maximum Bore Diameter Non Conformance			72.48 Log No.: SE00052	
D4-TC-3; 49/129;	ES199601368	Supplement 001	Revision 0000	Page 2 of 5

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the TC (Transfer Cask) top flange identified during TC fabrication.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Top Flange Maximum Bore Diameter Non Conformance			72.48 Log No.: SE00052	
D4-TC-3; 49/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the maximum bore diameter non conformance. The subject non conformance (Sulzer Bingham NCR No. 108834) identifies the maximum bore dimension of the TC cask top flange as 69.654" while the maximum allowable is 69.58". The oversize condition evidently resulted from shrinkage of the flange to shell weldment which caused an axisymmetric rotation of the flange about its centerline. The flange became slightly conical with an included angle of about 1 degree, so that it is slightly bell mouthed. The slight increase in maximum flange diameter will not affect the ability of the annulus seal to perform its function, and has no impact on any other cask design condition. The bore dimension is shown to be 69.55 +/- 0.03". Thus, the variance is only 0.074". Since the flange ring is 5.48" wide, this variance will not affect the annulus seal. Based on the above information, the subject non conformance will not affect the form, fit or function of the TC shell or the top flange, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC annulus seal to perform its intended design function. Additionally, the subject justification will not create any component assembly interference. Therefore, this activity has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the maximum bore diameter non conformance does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC

ATTACHMENT 3, SAFETY EVALUATION FORM

has not changed as a result of the maximum bore diameter non conformance, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the maximum bore diameter non conformance. The slight increase in maximum flange diameter will not affect the ability of the annulus seal to perform its function, and has no impact on any other cask design condition. The bore dimension is shown to be 69.55 +/- 0.03". Thus, the variance is only 0.074". Since the flange ring is 5.48" wide, this variance will not affect the annulus seal. Based on the above information, the subject non conformance will not affect the form, fit or function of the TC shell or the top flange, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC annulus seal to perform its intended design function. Additionally, the subject justification will not create any component assembly interference. Therefore, this activity has no detrimental impact on equipment important to safety, and does not create the possibility of a new malfunction.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity involved a maximum bore diameter non conformance. BGE approved this non conformance for construction prior to the issuance of the ISFSI license in November, 1992. The maximum bore diameter non conformance does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - TC Top Flange Maximum Bore Diameter Non Conformance			72.48 Log No.: SE00052	
D4-TC-3; 49/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the TC (Transfer Cask) top flange identified during TC fabrication.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Lead Pour Preheat Non Conformance **72.48 Log No.: SE00053**

D4-TC-4; 50/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 1 of 5**

Based on the attached discussion, does this activity:

Applicable to 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations

NO	Involve an unreviewed safety question (USQ)?
NO	Involve a change in the Technical Specifications/License Conditions or Bases?
NO	Require a change or addition to the UFSAR/USAR?

Applicable to 10 CFR 72.48 Safety Evaluations

NO	Involve a Significant Increase in Occupational Dose?
NO	Involve a Significant Unreviewed Environmental Impact?

Prepared by: J. E. Remeniuk
PRINTED NAME AND SIGNATURE
Department: NED-CEU 42-01-04 Date: 11-7-97

YES	Is a special review required by groups other than the group to which the Preparer belongs?

Resp. Ind.: G. Tesfaye Work Group: Licensing	Resp. Indv.: C. J. Dobry Work Group: PES	Resp. Indv.: R. H. Beall Work Group: NFM
11/12/97 SIGNATURE / DATE	11/10/97 SIGNATURE / DATE	11/11/97 SIGNATURE / DATE

[✓ Approved] Disapproved

Signature: Jon N Woodfield - Jon N. Woodfield
INDEPENDENT REVIEWER

Date 11/12/97

[✓ Approved] Disapproved

Signature: Michael J Gahan
for GS-DES, GS-TES, or PE-PDSU
MICHAEL J. GAHAN III

Date 11-13-97

The POSRC has reviewed this evaluation according to NS-2-101.

POSRC Meeting No.: 97-135 Date: 11·26·97

Recommend Approval ✓ Recommend Disapproval ________ Signature: ________ Date 11-26-97
POSRC CHAIRMAN

Approved ✓ Disapproved ________ Signature: ________ Date 12/1/97
PLANT GENERAL MANAGER

The OSSRC has reviewed this evaluation according to NS-2-100.

Full OSSRC Committee review required? Yes ________ No X

Signature: ________ Date: 1/30/98
OSSRC SES CHAIRMAN

If yes, OSSRC Meeting No.: ________

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Lead Pour Preheat Non Conformance **72.48 Log No.: SE00053**

D4-TC-4; 50/129; **ES199601368** **Supplement 001** **Revision 0000** **Page 2 of 5**

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the TC (Transfer Cask) shell identified during TC fabrication.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Function(s) of affected SSC: NUHOMS-24P (Nutech Horizontal Modular System) is a dry storage system that provides safe, interim storage for irradiated fuel assemblies. The system was designed by Pacific Nuclear Fuel Services (PNFS) (formerly Nutech Engineers, Inc.), which has since become Vectra Technologies, Inc. There are four major components of the NUHOMS-24P system. Those four components are 1) Dry Shielded Canister (DSC); 2) Transfer Cask (TC); 3) Lifting Yoke (Yoke); and 4) Horizontal Storage Module (HSM). A detailed description of each of these components is contained in the USAR and the NUHOMS-24P Topical Report. What follows is a brief description of the NUHOMS-24P system and those component(s) related to this evaluation.

NUHOMS-24P - the Calvert Cliffs license allows construction and operation of a total of 120 HSM's, which can house 2880 fuel assemblies. These modules can be built incrementally, as needed, to match BGE's requirements for additional storage. There are currently 48 HSM's constructed, which will allow for the continued generation and storage of spent fuel until approximately 2004. Each HSM contains one DSC, and each DSC contains 24 fuel assemblies. The fuel assemblies are transferred from the spent fuel pool via the DSC and the TC via the heavy haul road to the ISFSI site, where the DSC is then inserted into the HSM for interim storage.

Transfer Cask (TC) - the TC is a stainless steel cylinder with a bottom end closure assembly and a bolted top cover plate. There are two upper lifting trunnions near the top of the cask for downending / uprighting and lifting of the cask in the Auxiliary Building. The two lower trunnions serve as the axis of rotation during downending / uprighting operations and as supports during transport. The function of the TC is to provide radiological shielding during DSC closure operations and during transfer of the DSC to and from the ISFSI site. The TC is important to safety since it provides shielding and protection of the DSC from impact loads.

ISFSI USAR Revision No.: 5

ISFSI USAR Sections reviewed: The main chapters reviewed were 1, 3, 4, 5, 7, and 8. The key sections reviewed were 1.3, 3.3, 3.4, 3.6, 4.5, 4.7, 5.1, 8.1, and 8.2.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Lead Pour Preheat Non Conformance			72.48 Log No.: SE00053	
D4-TC-4; 50/129;	ES199601368	Supplement 001	Revision 0000	Page 3 of 5

Complete for 50.59 and 72.48:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

NO May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Probability of Malfunction:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as the result of this proposed activity. The NUHOMS-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The passive nature in itself provides a minimal probability for any malfunction to occur. There are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of the maximum shell preheat temperature non conformance. The subject non conformance (Sulzer Bingham NCR No. 109612) occurred during the preheat of the cask prior to the lead pour in which the area around the trunnions exceeded the maximum temperature of 725°F to a temperature of 880°F for approximately one hour. The shell material that experienced the temperature excursion is ASME SA Grade 304 with an actual carbon content of 0.058%. Per the Committee of Stainless Steel Producers of AISI, a time of 10 hours at a temperature of 500°C. (932°F) would be needed to form harmful amounts of chromium carbides. Since the actual temperature excursion was approximately one hour at 880°F, the time at temperature was insufficient to sensitize the material. The maximum temperature was observed about four inches from the trunnions. The actual ramp-up from 750°F to 880°F was quite rapid, about 15 minutes in duration, with an exposure of 30 minutes over 800°F and a total exposure of 1 hour and 50 minutes over 725 degrees F. It is not known what temperature was reached directly at the trunnion. It is known that the trunnion saw direct flame impingement during the 880°F temperature and that the high temperatures were only in the area of the trunnion. It is therefore likely that the trunnion was exposed to an even greater temperature. A sample was removed from the trunnion and tested for sensitization. The test confirmed that a condition of sensitization does not exist on the surface of the trunnion sleeve exposed to the elevated temperature. The material is therefore acceptable for use. The cask design is not otherwise affected by the temperature excursion. Based on the above information, the subject non conformance will not affect the form, fit or function of the TC shell or the trunnions, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform its intended design function. Therefore, this activity has no detrimental impact on equipment important to safety.

NO May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

Consequences of Malfunction:

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased as a result of this proposed activity. As stated above, there are no possible malfunctions of the TC which are described or evaluated in the USAR as a result of this proposed activity. As such, there are no consequences to consider.

NO May the probability of occurrence of an accident previously evaluated in the SAR be increased?

Probability of Accident:

The probability of occurrence of an accident previously evaluated in the SAR will not be increased as the result of the activity. One accident scenario described in the ISFSI USAR addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. The USAR states that an actual drop event is not credible, and the accident analysis concluded that fuel cladding integrity will be maintained for the postulated 80" transfer cask drop. Since the maximum shell preheat temperature non conformance does not adversely affect the ability of the TC to perform it's intended design function, the structural integrity of the TC is not affected, and as such, the probability of occurrence of the transfer cask accident previously evaluated in the SAR will not be increased as a result of this activity.

ATTACHMENT 3, SAFETY EVALUATION FORM

NO May the consequences of an accident previously evaluated in the SAR be increased?

Consequences of Accident:

The consequences of an accident previously evaluated in the SAR will not be increased as a result of this proposed activity. The cask drop analysis concluded that the transfer cask, the DSC, and its internal basket assembly and contained fuel will maintain its structural integrity through a cask drop. Since the intended design function of the TC has not changed as a result of the maximum shell preheat temperature non conformance, there will be no increase in the accident dose consequences already described in the USAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

NO May the possibility of a malfunction of a different type than any previously evaluated in the SAR be created?

Possibility of New Malfunction:

The possibility of a malfunction of a different type than any previously evaluated in the SAR will not be created as a result of this activity. The subject non conformance occurred during the preheat of the cask prior to the lead pour in which the area around the trunnions exceeded the maximum temperature of 725°F to a temperature of 880°F for approximately one hour. It is not known what temperature was reached directly at the trunnion. It is known that the trunnion saw direct flame impingement during the 880°F temperature and that the high temperatures were only in the area of the trunnion. It is therefore likely that the trunnion was exposed to an even greater temperature. A sample was removed from the trunnion and tested for sensitization. The test confirmed that a condition of sensitization does not exist on the surface of the trunnion sleeve exposed to the elevated temperature. The material is therefore acceptable for use. The cask design is not otherwise affected by the temperature excursion. Based on the above information, the subject non conformance will not affect the form, fit or function of the TC shell or the trunnions, is not detrimental to the structural integrity of the TC, and will not adversely affect the ability of the TC to perform its intended design function. Therefore, this activity has no detrimental impact on equipment important to safety, and does not create the possibility of a new malfunction.

NO May the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Possibility of New Accident:

The possibility of an accident of a different type than any previously evaluated in the SAR will not be created as a result of this activity. No new accident scenarios are created as the result of this proposed activity.

Complete for 50.59 and 72.48:

3. The margin of safety as defined in the basis for any ISFSI Technical Specification is not reduced.

NO Will the margin of safety as defined in the basis for any ISFSI Technical Specification be reduced?

Bases Discussion of why the margin of safety is not reduced

None of the Technical Specifications nor the Bases are affected by this activity.

Complete for 72.48:

NO Will the proposed activity involve a significant increase in occupational dose?

A significant increase in occupational dose:

A significant increase in occupational dose will not occur as a result of this proposed activity. The activity involved a maximum shell preheat temperature non conformance. BGE approved this non conformance for construction prior to the issuance of the ISFSI license in November, 1992. The maximum shell preheat temperature non conformance does not adversely affect the operation or the associated occupational exposures as described in ISFSI USAR Table 7.4-1.

ATTACHMENT 3, SAFETY EVALUATION FORM

ISFSI - Transfer Cask Lead Pour Preheat Non Conformance			72.48 Log No.: SE00053	
D4-TC-4; 50/129;	ES199601368	Supplement 001	Revision 0000	Page 5 of 5

NO Will the proposed activity involve a significant unreviewed environmental impact?

A significant unreviewed environmental impact:

A significant unreviewed environmental impact will not occur as the result of this proposed activity. The proposed activity does not affect the environmental conditions of the ISFSI.

Summary: (For NRC Report, provide a brief overview)

Proposed Activity: To evaluate an ISFSI non conformance that occurred prior to the issuance of the ISFSI license in November, 1992. This particular safety evaluation addresses a non conformance with the TC (Transfer Cask) shell identified during TC fabrication.

Reason for Activity: This non conformance was fully evaluated and justified by Pacific Nuclear Fuel Services and approved by BGE for construction prior to the issuance of the ISFSI license in November, 1992. This was included in a document which was submitted to the NRC on July 16, 1992, which provided the first revision to the original SAR and provided changes made to ISFSI design documents during fabrication that had not been previously reviewed by the NRC. This safety evaluation was performed because the NRC has not reviewed that submittal.

Activity Summary: After a thorough and intense review, it has been concluded that the ISFSI documentation reviewed:

- Does not constitute an Unreviewed Safety Question (USQ)
- Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR
- Does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR
- Does not reduce the margin of safety as defined in the basis for any ISFSI Technical Specification
- Does not result in a significant increase in occupational dose
- Does not constitute an Unreviewed Environmental Impact (UEI)